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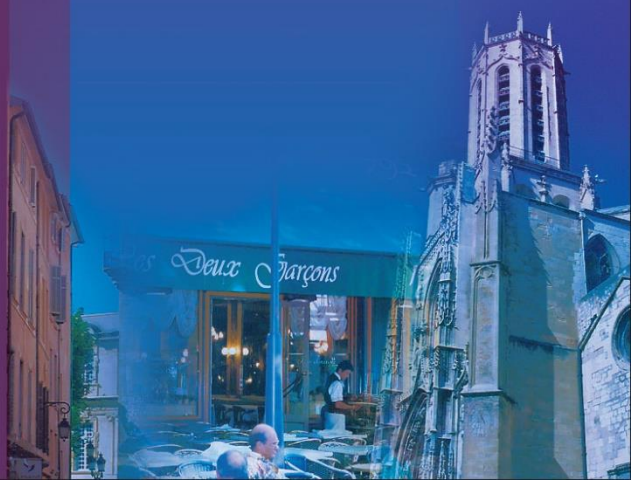
PROCEEDINGS

Aix en Provence
France
22-24 May
2007

# 29th Annual Meeting

# Symposium on Safeguards and Nuclear Material Management




ESARDA
EUROPEAN SAFEGUARDS RESEARCH AND DEVELOPMENT ASSOCIATION

# ESARDA 29th Annual Meeting

## Symposium on Safeguards and Nuclear Material Management

## Aix en Provence, May 22-24, 2007

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# Plenary session

# Opening of the Symposium

Liberté • Égalité • Fraternité
RÉPUBLIQUE FRANÇAISE

**Ministère de l'Economie, des Finances et de l'Industrie**
**Haut Fonctionnaire de Défense et de Sécurité**

# LA PROTECTION ET LE CONTRÔLE DES MATIÈRES NUCLÉAIRES EN FRANCE

ESARDA
22 mai 2007

MINISTÈRE DE L'ÉCONOMIE
DES FINANCES ET DE L'INDUSTRIE

# Le contexte général (1)

- Depuis 1945 la France a développé un programme nucléaire de grande ampleur :
  - civil et militaire (78 % de l'électricité nucléaire)
  - couvre l'ensemble du cycle du combustible nucléaire
  - nombreuses installations d'essai et de recherche
- Toutes ces installations mettent en jeu des matières fissiles et fertiles
- l'Etat prend en compte le risque associé et se sent responsable tant vis-à-vis des citoyens que de la communauté internationale

# Le contexte général (2)

- Le ministre de l'Industrie est responsable, avec l'appui de l'IRSN de l'application de la réglementation relative aux transports de matières nucléaires et aux installations nucléaires
- Responsabilité première des opérateurs qui doivent connaître en permanence la quantité, la qualité et la localisation des MN qu'ils détiennent

# Le contexte général (3)

- Historiquement, le CEA, établissement public de l'Etat, seul propriétaire de MN en France
- Evolution économique (privatisations, apparitions de nouveaux acteurs [AREVA, EDF])
- Nécessité de prendre en compte la menace terroriste et de l'évolution des textes internationaux

# La protection et le contrôle des MN (1)

- Objectif : protéger les MN
- Mesures pour répondre à cet objectif :
  - suivi et comptabilité des MN
  - mesures de protection physique des installations et des transports
- Ces mesures se déclinent selon 5 axes :
  - prévention, détection, alerte, réaction, limitation des conséquences

# La protection et le contrôle des MN (2)

- Mise en œuvre du contrôle national des MN sous responsabilité du HFDS auprès du MINEFI
  - BCMNS (10 personnes) : animation et coordination des actions de protection et de contrôle
  - appui technique de la DEND de l'IRSN (± 100 personnes)
    - inspection dans les installations et en cours de transport
    - analyse des dossiers fournis par les titulaires d'autorisations
    - suivi des transports de MN
    - comptabilité centralisée des MN

# La protection et le contrôle des MN (3)

- Importation, exportation, élaboration, détention, transfert, utilisation et transport de MN soumis à autorisation préalable du HFDS
- Autorisation assortie de conditions (durée, quantité et forme physico-chimique des MN, suivi et comptabilité des MN)
- En dessous d'un seuil variable selon les matières, régime de simple déclaration annuelle des matières détenues et des activités exercées

# La protection et le contrôle des MN (4)

- Certains agissements portant sur les MN sont des délits correctionnels assortis de sanctions parfois lourdes (jusqu'à 10 ans de prison) : appropriation indue de MN, exercice sans autorisation d'activités touchant aux MN, fourniture de fausses informations pour obtenir une autorisation, ...

# La protection et le contrôle des MN (5)

- Au 1er niveau contrôle exercé par le HFDS sur respect des conditions, sur le suivi et la comptabilité et dispositions destinées à prévenir le vol ou le détournement de MN
- Au 2ème niveau inspections par inspecteurs des matières nucléaires sur la manière dont les titulaires d'autorisations appliquent la réglementation et respectent leurs engagements

# Les engagements internationaux de la France

- France soumise au contrôle de l'AIEA et d'EURATOM
- Ces contrôles et le contrôle national assurent une protection efficace des MN
- France puissance nucléaire civile et militaire, mais ne dispose pas de 2 cycles séparés pour fabrication des MN
- D'où dispositions permettant à la fois :

- Le contrôle exhaustif des MN civiles ;
- La préservation de l'indépendance nationale en matière de défense,
- La fourniture des données comptables à l'AIEA et à EURATOM ;
- La conduite des inspections de l'AIEA et d'EURATOM

# Conclusion (1)

- Sécurité nucléaire en général, protection et contrôle des MN, des installations et des transports en particulier, enjeu majeur tant en terme de lutte contre la prolifération que contre le terrorisme
- Dispositif législatif et réglementaire français répond à cet enjeu
- Pleine participation des opérateurs français, conscients de leurs responsabilités
- A court terme, le gouvernement français proposera au parlement la ratification de l'amendement à la CPPMN adopté par l'AIEA en juillet 2005, dont il respecte déjà les dispositions essentielles
- Participation active de la France à la révision de l'INFCIRC 225

# Conclusion (2)

- Ces différents éléments constituent des jalons essentiels pour la sécurité et la défense économique de la France
- Ils s'inscrivent dans la durée et sont cohérents avec l'importance que revêt l'énergie nucléaire en France
- En tant qu'acteur nucléaire majeur dans le monde, la France se sent aussi tenue d'être exemplaire en matière de sécurité nucléaire

# Contributions of ESARDA and JRC to address new challenges for nuclear safeguards, non-proliferation and nuclear security.

*R. Schenkel*
*European Commission Joint Research Centre*
*Director General,*
*Brussels*

http://www.jrc.cec.eu.int

# Outline

- ESARDA successfully addresses current needs through its working groups:
  - Training and Knowledge Management
  - NMAC Audit Focus
  - Import/Export Control (in WG Verification Technologies/Methodologies)
- JRC pursues new developments in nuclear safeguards & security to support the Commission, Member States and IAEA in collaboration with international partners
- Challenges for safeguarding future nuclear systems
- FP7 Nuclear and Security Programs : priorities and budget
- International collaboration extending towards security
- Conclusions

# Replying to the nuclear safeguards education needs



• ESARDA WG TKM picked up successfully the challenge to address the need for nuclear safeguards and non proliferation EDUCATION at university level

• A 3rd international course took place in Ispra, March 2007, with over 60 students from 18 nationalities and high level lecturers (incl. IAEA, AREVA, US lab…)

•The course content and material is based upon input and validation of ALL ESARDA WG's and input from other international. experts

• IAEA uses this expertise for a planned International Coordination of Education and Training in Nuclear Security and Non-proliferation (Vienna, April 2007)

• Next years ESARDA course session is scheduled for March 24-28, 2008

• Remaining challenge is to establish a full-fledged course syllabus for long term academic recognition (by ENEN : European Nuclear Education Network)

# Combining audit and inspection approaches



- ESARDA WG Audit Focus picked up successfully the challenge to address the audit focus of the DG TREN initiative to introduce a new safeguards approach
- This was in line with priorities of the COREPER Working Party on Atomic Questions and other international initiatives for best practices on security and NMA
- The guidelines identify detailed criteria, quality assurance issues and good practice approaches for accountancy and control of NM. They emphasize the opportunities for flexibility in the means used by facilities for meeting regulatory requirements,
- They put in evidence the high level of expertise currently being deployed by EU facilities and enable sharing of expertise with new member states,
- The JRC will continue working with ESARDA and DG TREN for developing and testing audit methodologies in NMAC

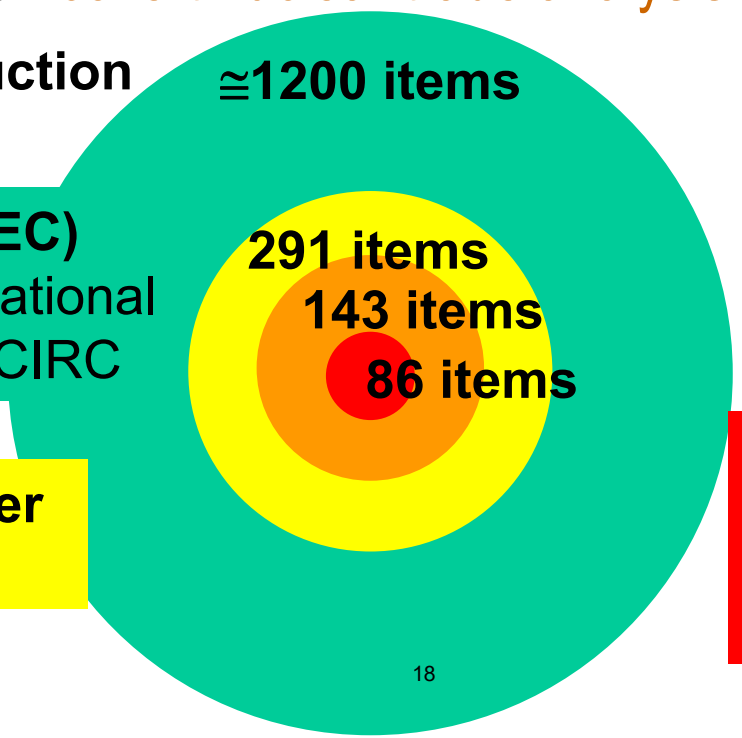
# Verifying import/export of dual-use items/technologies



• ESARDA WG Verification Technologies and Methodologies organized a first intl. workshop in Nov 2006 to address the dual use technology issues and related import/export control

•JRC Activities in this field include
- Correlation table between trade common nomenclature and trigger lists
- Use of language technology for automation of searches
- IAEA-EC SP task on covert nuclear trade analysis (incl. Contraffic data)

**Example of data reduction**




**Council Regulation (EC)**
Integrating most international available lists and INFCIRC

**INFCIRC 254 : Trigger List (IAEA, NSG)**

- **Reprocessing**
- **Separation of uranium isotopes**
- **Heavy water production**

- **Reprocessing**
- **Subset of separation of U isotopes (CF + laser)**
- **Heavy water production**

# Selected current JRC developments in SFG & Security

- Novel Verification Technologies with γ-ray Imaging with 3-D LIDAR
- Detailed geographically referenced and hierarchically structured information from very small (micron) to very large (km) range (incl. GIS, CAD etc…)
- Single particle analysis and sizing system (SPASS) : new applications
- Miniaturized Capillary Electrophoresis e.g. for measuring TBP traces
- A combined CAlorimetry, NEutron coincidence counting and GAmma spectrometry system (CANEGA) for enhanced plutonium mass and isotopics
- Training in additional protocol / complementary access with IAEA/DG TREN
- New digital-based acquisition systems for neutron multiplicity counting intended to replace in the medium term the traditional analogue electronics
- Intl. collaboration in applied nuclear security developments and support programmes : Border Monitoring Working Group and Nuclear Forensics
- Discrimination between Naturally Occurring Radioactive Materials and man-made radioactive sources and nuclear material

# Novel Verification Technologies

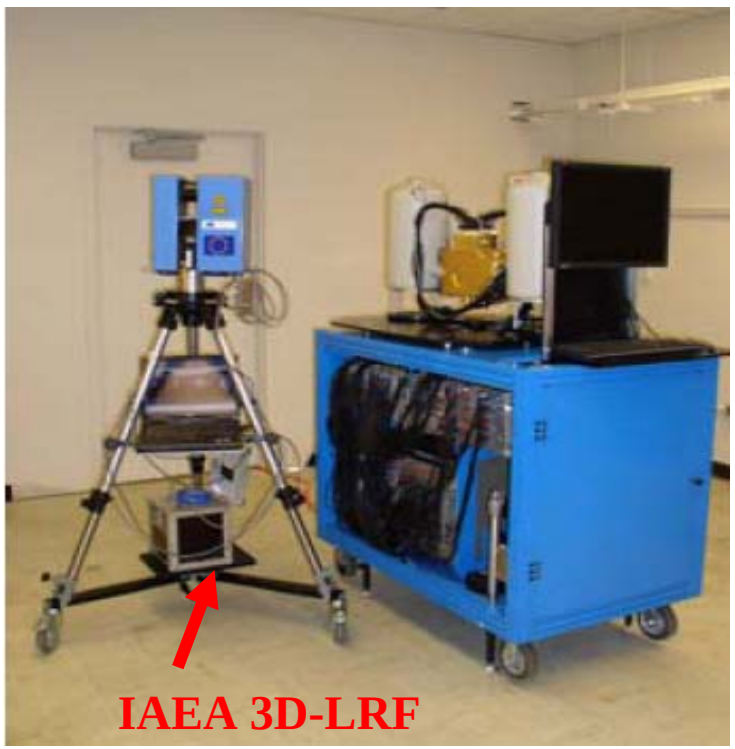
## γ-ray Imaging with 3-D LIDAR

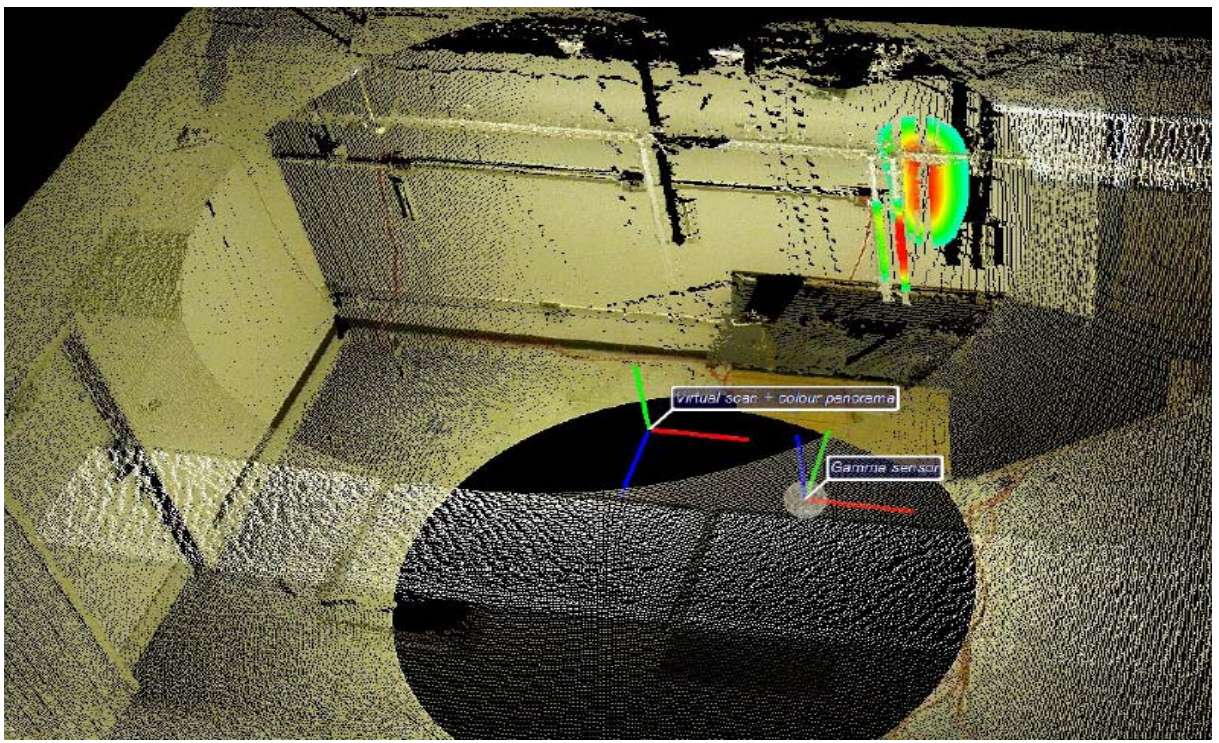
**Source: JRC/LLNL/ORNL**



**Perceived need:** Design information and nuclear materials distribution verifier for complex nuclear facilities

**Novel features:** Combines JRC's 3D DIV system (laser scanner and software) with a LLNL-developed Compton γ camera. Successful joint tests made with US Oak Ridge and Lawrence Livermore National Laboratories.




**Remark(s):** **Possible other applications in the verification of process hold-up in difficult-to-access areas and cascades**

# Single Particle Analysis and Sizing System SPASS: Schematic Overview

For each particle, the full mass spectrum (pos and neg) is recorded

→ quick screening of particle composition on a swipe sample (e.g. before SIMS)

→ identify characteristic elemental signatures & indicators (Ba, Cd, Li, Pb)

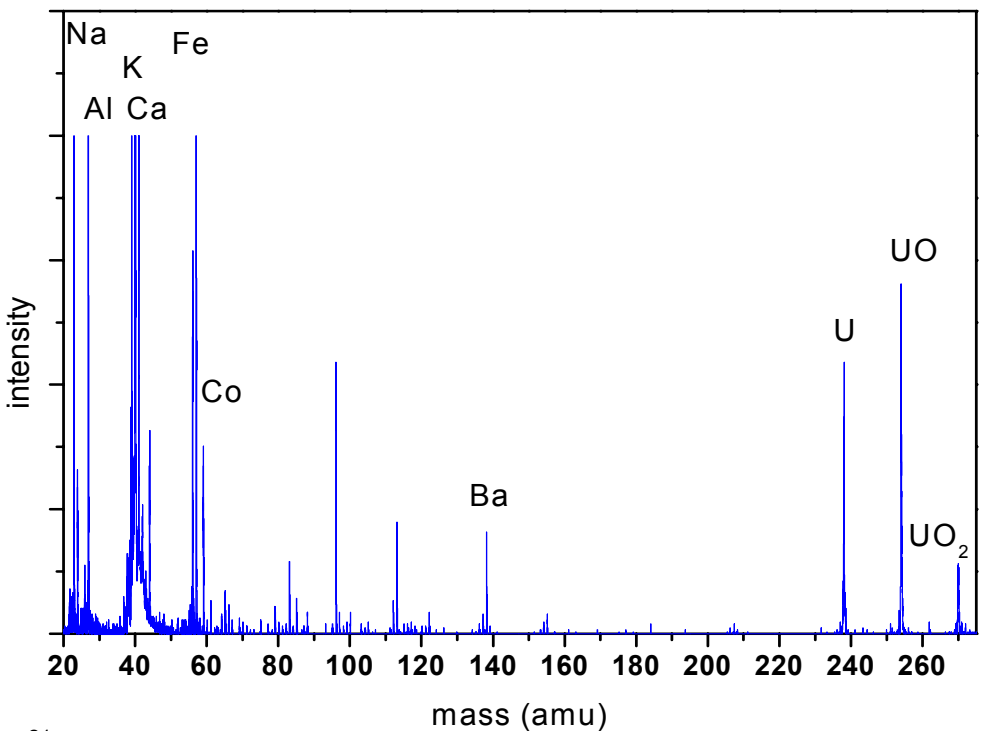
Impurity analysis can enable confirmation that:

-Processes declared are those used

-Purity levels are consistent with processes used

- Compounds produced at one facility and used at another are the same

- Only declared (as to source) materials are present at a facility



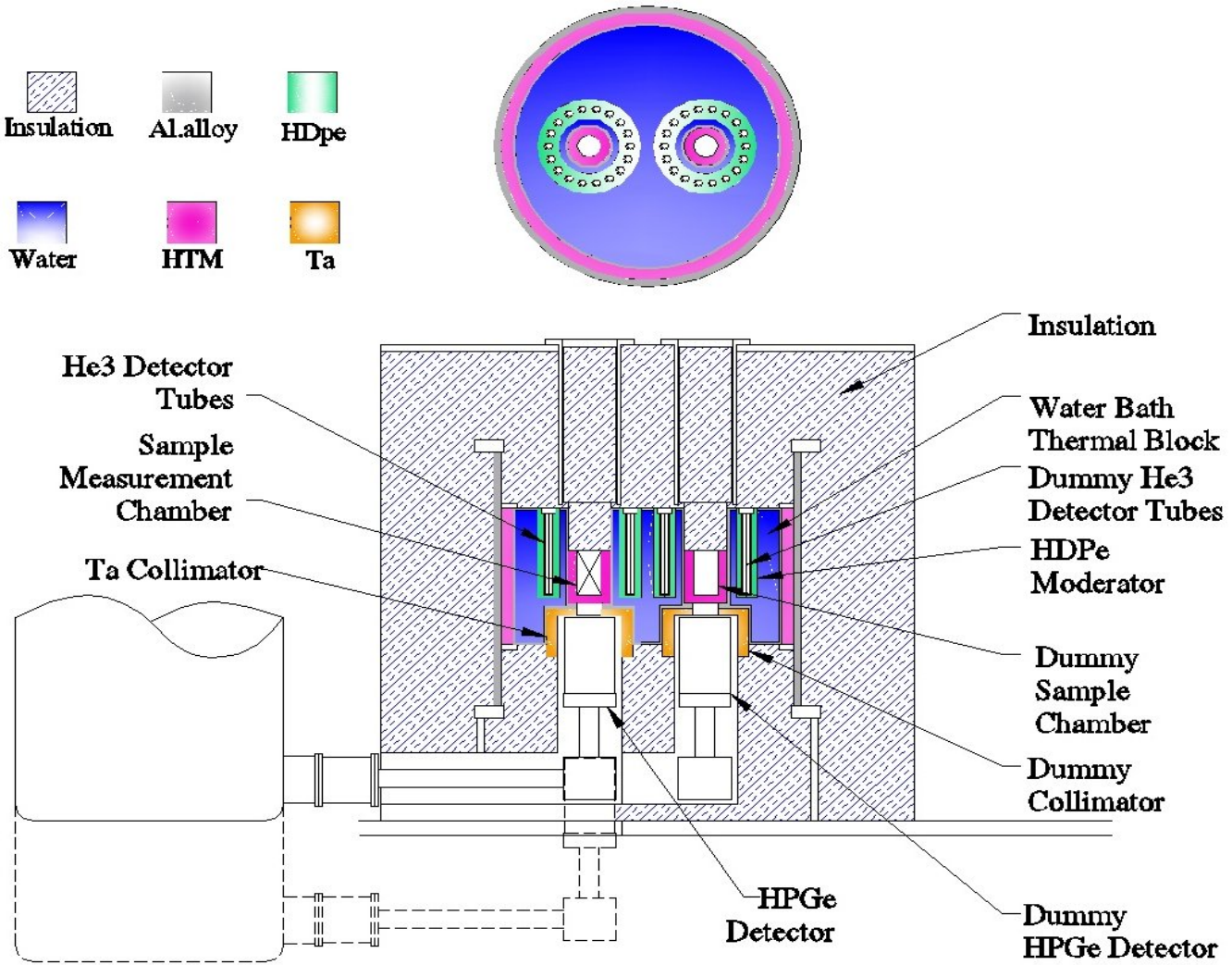
# Study on Combined CAlorimetry, NEutron coincidence counting and GAmma spectrometry system (CANEGA) for enhanced Pu determination



**Objective:**

Allows to overcome the Pu-242 problem and

Allows to detect eventual effects of presence of Cm isotopes



# Training in additional protocol / complementary access



- Collaboration with both IAEA and DG TREN

  First course, Ispra 03/07

  3 teams of 4 inspectors

- Complementary access
  - spent fuel pond
  - reactor
  - hot cells
  - tritium laboratory

- The training allows to test and improve the investigative skills and to focus on the observational, communication, negotiating, and team building skills currently required of nuclear inspectors in the detection of undeclared activities.

# Intl. collaboration in applied nuclear security: Border Monitoring Working Group & Nuclear Forensics



## BORDER MONITORING

- International Working Group with US DOE / IAEA / EU
- R&D on NORMS and detectors
- Prototype of a radiation source for portal monitor testing
- Future : ITRAP+10 with SAL

## NUCLEAR FORENSICS

- International Techn.Working Group
- R&D on fingerprinting/particles
- Training courses for responsible services on forensic techniques
- Sample analysis upon request e.g. 2 recent German cases 2006/2007


# Challenges for safeguarding future reactor systems



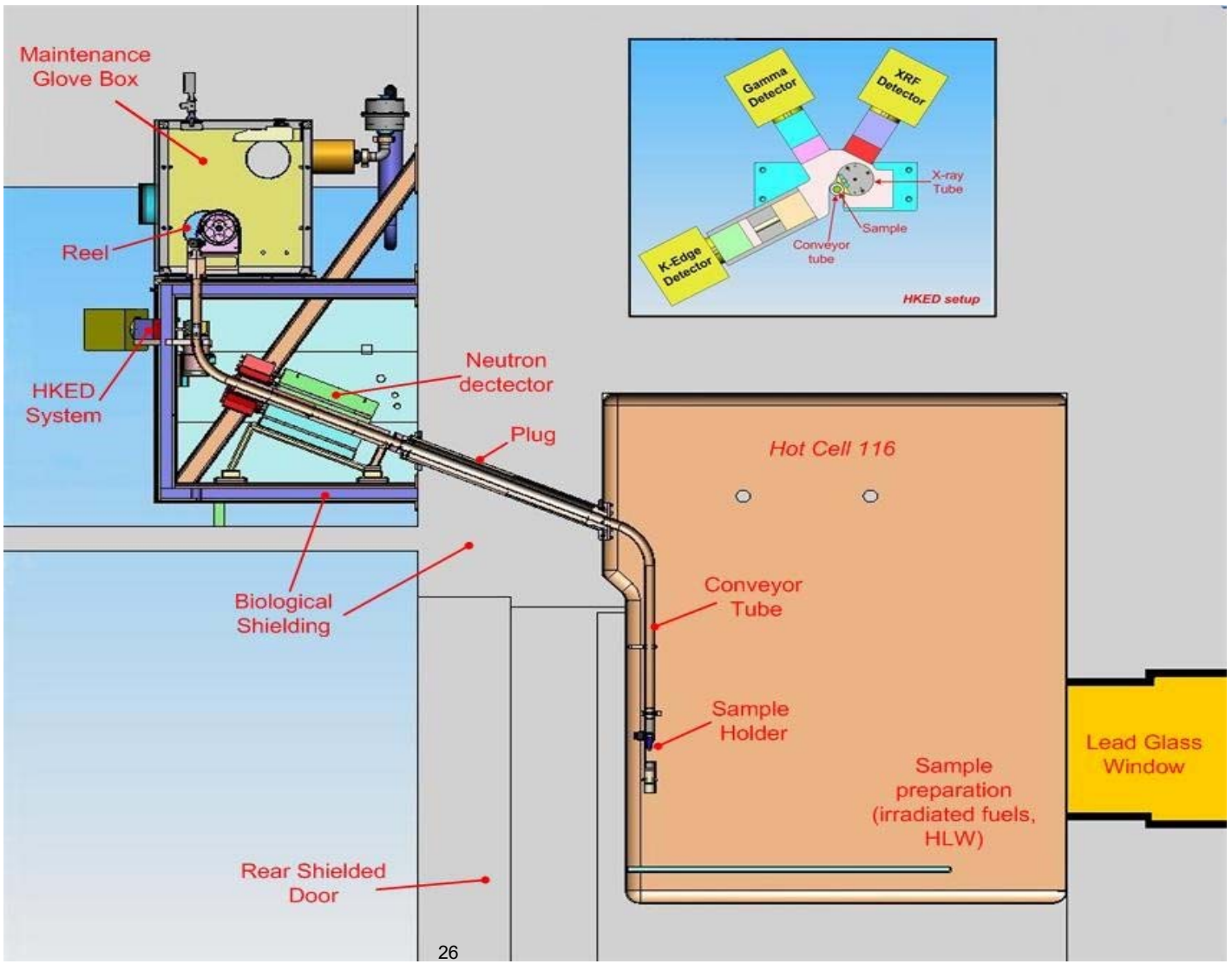
- Development of methodologies for Proliferation Resistance and Physical Protection (PR&PP) evaluation in the frame of Generation IV Int. Forum
  - PR&PP stable paradigm, based on threat characterisation, pathway analysis and estimation of PR&PP features, issue of Rev. 5 methodology approved by GIF EG
  - Presented to Gen IV systems designers in USA, Europe, Japan;
  - Tested on a simplified GIV nuclear energy system (Pyroprocessing facility associated to an example sodium fast reactor) using qualitative, Markov and Event/Fault tree approach
  - Needs closer collaboration with designers and more comprehensive applications
- Enhancing Safeguardability of new concepts via:
  - Development of methods for analysing Safeguardability of future nuclear systems at various design stages, aimed at providing feedbacks and guidance for improvement to systems designers;
  - Develop innovative analytical measurements for advanced fuel concepts (see next slide)
- Application of Event Tree / Fault Tree technique to the modelling/analysis of nuclear proliferation and nuclear security scenarios
  - Investigation of the potentialities of non-coherent FT modelling of nuclear proliferation and nuclear security scenarios;
  - Development of JRC ASTRA-FTA software for exact analysis of non-coherent FT relevant for security applications.

# Multi-purpose NDA station (KED, XRF-Ge, XRF-Si, NCC, HRGS) for direct analysis of fuel specimens from pyro-processing

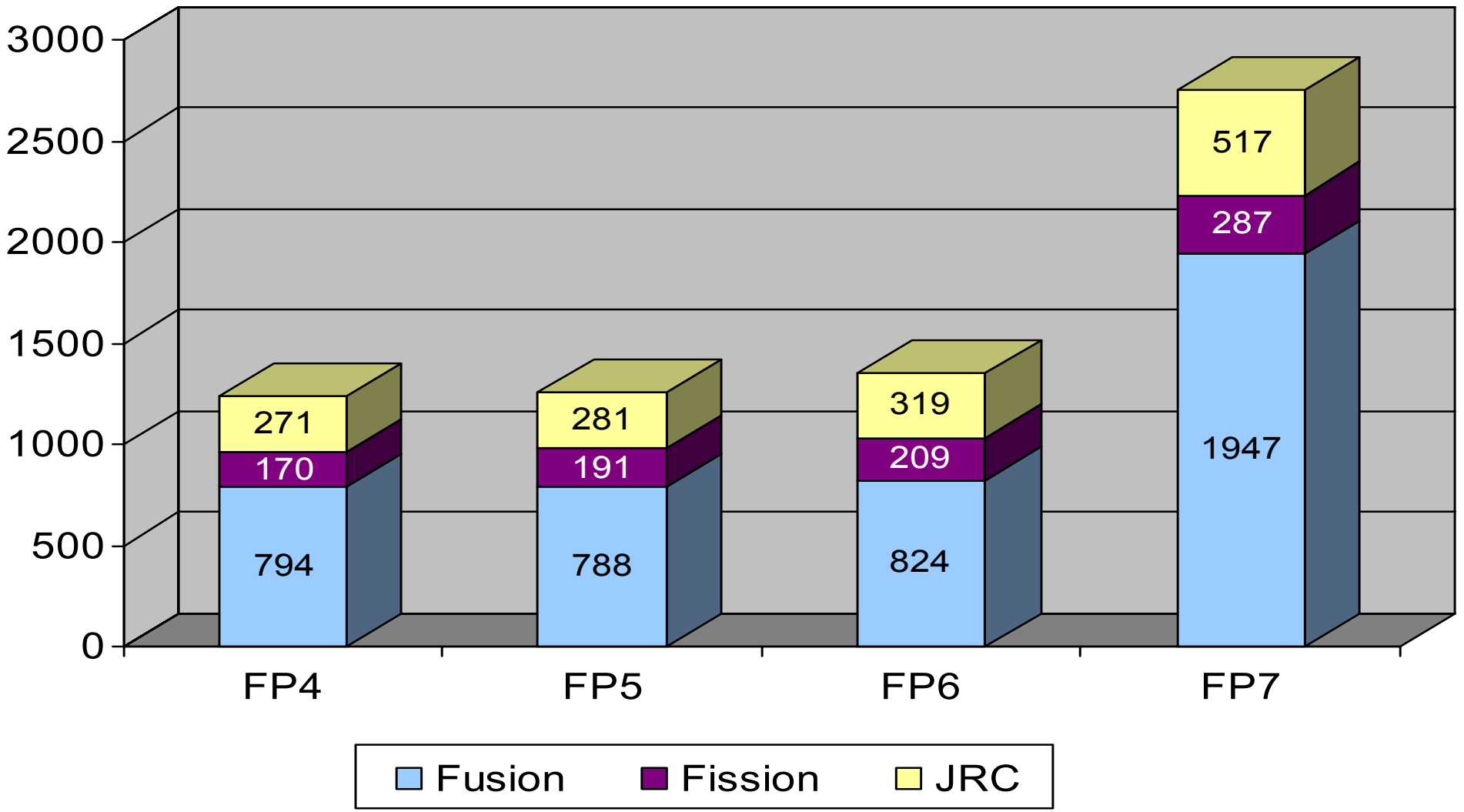


**Objective:**

Provide measurement capabilities for **direct** Lanthanide / Actinide analysis (the latter at concentration levels ( > 0.1 g/l))



# EURATOM FP7 Research budget (€ Million)



# Energy Research in FP7
## EURATOM
## nuclear fission & radiation protection

**Geological disposal of long-lived radioactive waste**

**Reducing toxicity of radioactive waste through partitioning & transmutation**

**Potential of innovative reactor concepts**

**Operational safety of existing and future reactor systems**

**Risks from exposure to ionising radiation – especially medical uses and low doses**

**Radiological terrorist threats**

**Support for infrastructures and the retaining of competences and know-how in all areas of nuclear science**

# Security FP7 Activities and budget



4 mission areas

- Security of citizens (technology solutions for civil protection, bio-security, protection against crime and terrorism)
- Security of infrastructures and utilities (examining and securing infrastructures in areas such as ICT, transport, energy and services in the financial and administrative domain)
- Intelligent surveillance and border security (technologies, equipment, tools and methods for protecting Europe's border controls such as land and coastal borders)
- Restoring security and safety in case of crisis (technologies and communication, co-ordination in support for civil, humanitarian and rescue tasks)

3 Cross cutting activities:

- Security systems integration, interconnectivity and interoperability (information gathering for civil security, protection of confidentiality and traceability of transactions)
- Security and society (acceptance of security solutions, socio-economic, political and cultural aspects of security, ethics and values, social environment and perceptions of security)
- Security research co-ordination and structuring (co-ordination between European and international security research efforts in the areas of civil, security and defence research)

Security Research in FP7: Budget 2007-2013: ~1,400 M€

Work Programme 2007 based on budget 2007 + 2008 => ~170 M€

~ 150 M€ for Security Research Call 1 and ~ 20 M€ for Coord. Call Security-ICT

# International collaborations extending from nuclear safeguards and non-proliferation towards security



## IAEA-EC SP

+ Very solid support programme ( 25 years) covering wide range of issues in nuclear safeguards and non-proliferation (over 30 tasks)

+ Good information sharing with other Member States Support Programmes (incl joint meetings)

+ Examples of tasks recently approved :
  - Software Engineering Support for 3D Camera Development
  - Software, Hardware and Database Provision for Satellite Imagery Analysis Support
  - Signatures of Nuclear Fuel Cycle Related Processes
  - Novel Techniques and Instruments for Detection of Undeclared Nuclear Facilities, Material and Activities
  - Use of Satellite Imagery Data for Geological Repositories Monitoring

+ Tasks linked to nuclear security e.g. Open Source Info Collection / Covert Nuclear Trade Analysis

+ Additional collaboration with IAEA Nuclear Security Department (e.g. combating illicit trafficking)

# International collaborations extending from nuclear safeguards and non-proliferation towards security

US DOE – EURATOM agreement : 11 running tasks and 8 new task sheets under preparation

e.g. - Characterization of the Capture-Gated Liquid Scintillator BC-523

- Investigation of Combined Measurements with 3D Design Information Verification System, Gamma-Ray and Neutron Imaging Systems for International Safeguards Applications
- Proliferation Resistance and Physical Protection Evaluation Methodologies
- Reference material for Uranium Age Determination

Agreement under revision to open up towards nuclear security and more info sharing with IAEA

US DOE NNSA incl. IAEA : request for integrated training for Second Line of Defence Activities

CANADA incl. DG TREN : very active in the PR&PP (Canada co-chairing GIF Expert Group)
New : sharing experience for CANDU safeguards assessments (Workshop 29-30/05/07 Ispra)

RF and CIS via DG RELEX : important TACIS projects running (30 Meuro / 5 years) in area of Nuclear safeguards, non-proliferation and nuclear security (e.g. Multicountry project on illicit traffic.)

GEOGRAPHICAL EXTENSION via DG RELEX : Global Stability Instrument : incl threats of WMD (incl. illicit trafficking, dual use, emergency response) with +/-300 Meuro for the period 2007 – 2012

# Conclusions

- ESARDA has demonstrated to be able to efficiently address new challenges in nuclear safeguards and related matters
- Many European and International initiatives are running and/or starting in the field of radiological and nuclear threat assessments, threat reduction and response capabilities, which rely upon the expertise and skills of the nuclear safeguards community
- ESARDA should be aware of and continue to exploit its influential and important role in gathering nuclear security competences in Europe (Conclusion also formulated in London 2005, but now more true then ever)
- JRC is a reliable partner in safeguards and non-proliferation R&D and is looking forward to strengthen also forthcoming international relationships in nuclear security applications
- JRC will continue to make its skills and laboratories available to DG TREN, IAEA and the international nuclear safeguards and security community

# Safeguards – Need for New Verification Technologies

**Olli J. Heinonen**
**International Atomic Anergy Agency**
**Deputy Director General, Head of the Department of Safeguards**

This year´s *ESARDA Annual Meeting* has special significance for the Safeguards Community because we are celebrating a number of significant anniversaries: 50 Years of the *Euratom Treaty*, 25 years of the French Support Programme and 50 Years of the IAEA, 50 Years of Atoms for Peace. I would like on behalf of the Director General and the staff of the IAEA to express sincere appreciation for the collaboration and support from Euratom and the French Support Programme in our work. The IAEA relies heavily on Member State support, without which our task would be unimaginably more difficult. So, we mark this anniversary celebration as a commitment to the future, to continued collaboration and support and in particular to the implementation of Integrated Safeguards in all NNWS of the EU.

It is also a pleasure for me to attend this Annual Meeting because it provides an opportunity for an exchange of information and experience on many aspects of international safeguards and nuclear non-proliferation. This for me, as Head of the Department of Safeguards, is an exciting arena because it gives me and my staff a glimpse at a number of tools that may be beneficial, or even essential, to our work in the future. In turn, some of my staff will be presenting parts of our work, and I hope that we can manage to equally spark your interest and enthusiasm.

Safeguards and nuclear material management and accountancy have changed dramatically over the years - I believe we are all we aware of this. The challenge for all of us in the continuous race against nuclear proliferation is to address these changes as quickly and effectively as possible. We also endeavour to monitor future trends in order to minimize as far as possible any surprises, and to ensure that we remain adaptable and able to respond decisively. The Research and Development Programme for Nuclear Verification in the IAEA is a key element of our efforts to "stay ahead of the game" and ensure that we are as well equipped as possible to face future challenges.

The Department of Safeguards is currently pursuing, among others, R&D projects in the areas of:

1. Enhanced information analysis architecture and understanding nuclear trade mechanisms;

2. Novel technologies, including NDA techniques;

3. Unattended Remote Monitoring Systems.

One of the major shifts since the commencement of Safeguards has been the collection, analysis and evaluation of information. Collection, analysis and evaluation of information is an area that is experiencing a "renaissance" in Safeguards, and will continue to evolve in the years to come. Information is at the heart of modern verification; in fact, we frequently refer to our work as being *information-driven safeguards*.

The *safeguards conclusions* that we draw every year for each State with a safeguards agreement in force are *based explicitly on the evaluation of all information available* to the Agency. However, with current resources and techniques there are clear limits to how much information can be evaluated - and the amount of information available is constantly growing. New technologies and well-trained staff can offer real solutions for the collection and analysis of these increasing volumes of information.

As we look even further into the future, beyond the early years of the next decade, we can see that *proliferation resistance* issues will require consideration. Although we have begun to explore better control of access to nuclear fuel cycle technology through initiatives such as multinational approaches to parts of the nuclear fuel cycle, and ensuring reliable supply of reactor fuel, *novel technologies* will be an additional, and essential, component of safeguards work in these areas. Advancements such as INPRO, GEN IV and fast reactors will also need to be monitored with a view to their needs for safeguards.

We are also examining a broad range of new technologies for their applicability in nuclear material verification. These include: laser induced breakdown spectrometry, noble gas analysis, light detection and ranging for monitoring gaseous atmospheric pollutants, optical stimulation luminescence, and semiconductor sensors for $UF_6$, (uranium hexafluoride).

The IAEA does not have the means to perform its own R&D and it relies solely on Member State support in this regard. We have received, for example, invaluable support from the French Support Programme in the areas of satellite imagery, information analysis, staff training and in particular in the development of equipment. As we celebrate the many years of cooperation from both Euratom and the French Support Programme, we look forward to their continued commitment to supporting us in providing the international community with the best possible assurances about the peaceful use of nuclear energy.

We can also benefit from the support of organizations, such as those here this week, capable of helping us in the development of advanced technologies that will provide effective solutions to verification challenges in the future.

Our objective at the IAEA is to continue to verify, in an impartial and independent manner, that States are honouring their safeguards obligations. In this regard we are always looking to, and searching for, any tools and approaches that will make our work more effective and efficient: I have no doubt that the extensive scope of this Annual Meeting will give us much to consider. Thank you.

*This text is also available at the following URL*
*http://www.iaea.org/NewsCenter/Statements/DDGs/2007/heinonen22052007.html*

# Euratom and the role of nuclear in European Energy Policy

*by Dominique Ristori, Deputy Director-General,*
*Directorate-General for Energy and Transport, European Commission*

## Executive summary

1. Member States in the European Union are facing the three following **key energy challenges**:
   - *Climate change*: energy amounts to 80% of all greenhouse gas emissions in the EU;
   - *Security of supply*: Europe is becoming increasingly dependent on imported hydrocarbons: reliance on imports is expected to increase by 2030 to 84% for gas and 93% for oil. In a context of spectacular growth of global demand (increase by 60% by 2030), pressure on world energy resources is intense.
     At the same time EU electricity demand is rising by 1,5% per year. Even with an effective energy efficiency policy, investment in generation alone over the next 25 years will be necessary in the order of €900 billion within the EU;
   - *Competitiveness*: the EU is becoming increasingly exposed to the effects of price volatility and price rises on international energy markets.

2. At the European Council of 8 and 9 March 2007 the European Heads of State and Government took note of the European Commission's assessment of the **contribution nuclear energy can make** in meeting the above key challenges:
   - *Climate change*: Nuclear is one of the largest sources of $CO_2$ free energy in Europe. For those Member States that wish, it is likely to form part of an energy scenario where significant emission reductions are going to be required in the coming decades;
   - *Security of supply*: Currently around one third of the electricity in the EU comes from nuclear. Nuclear energy contributes to diversification and long term security of supply for the following reasons:
     - The sufficient availability and wide geopolitical distribution of uranium resources and suppliers: Australia and Canada currently support 45% of the EU uranium requirements;
     - The limited importance (10 to 15%) of the raw material – natural uranium – in the total cost of generating electricity;
     - The existing production capabilities.
   - *Competitiveness*: Nuclear power is one of the cheapest sources of low carbon energy. It is less vulnerable to fuel price changes than coal or gas-fired generations, as uranium represents a limited part of the total cost of generating nuclear electricity and is based on sources which are sufficient for many decades and widely distributed around the globe.
     The nuclear sector is also one of the areas where the EU has the world technological leadership (in particular for new nuclear reactors, enrichment and reprocessing).

3. For 50 years the **Euratom treaty has provided a stable and reliable legal framework** for the development of nuclear energy in Europe. The Euratom treaty has provided :

   - a high level of safety and security, including in the context of enlargement;
   - the basis for supervising the use of nuclear materials for peaceful purposes in the Community;
   - a high level of radiation protection for all EU citizens by laying down basic standards and monitoring compliance;
   - an important contribution to scientific progress by supporting nuclear research;
   - equal access to nuclear resources for all users;
   - strengthened international cooperation in the nuclear field.

4. In line with the conclusions of the European Council of 8 and 9 March, the role of the European Union should be to **develop further, in conformity with Community law, the most advanced framework for nuclear energy. Priority actions under the Euratom Treaty** are focussing in particular on the following topics :

   - *Non proliferation and cooperation with the IAEA*: One of the key priorities of the external EU energy policy is to promote at international level the highest standards of non proliferation, nuclear safety and security. In this context the Commission called for a reinforced cooperation with the International Atomic Energy Agency (IAEA). This closer cooperation is already effective since several months and could be formalized in the following months in a new Charter of cooperation between the IAEA and the Commission.

   - *Nuclear safety and waste management:* With the support of the European Council and the European Parliament, the Commission will soon establish a High Level Group on nuclear safety and waste management with the mandate of progressively developing common understanding and, eventually, additional European rules, on nuclear safety and waste management. The EU also continues its efforts in the field of research by launching on 21 September 2007 the European Sustainable Nuclear Energy Technology platform.

   - *Nuclear energy forum* : The EU Nuclear Energy Forum is one of the initiatives taken by the Commission to improve the methods of European governance through greater transparency and reinforced dialogue between the European Commission and interested parties. In line with the European Council's conclusions, it will provide a platform for structured dialogue with all relevant parties – both institutions and stakeholders – to discuss relevant nuclear issues. The Forum will be organized alternatively in Bratislava and Prague following an agreement reached between the two Prime Ministers concerned and the European Commission. The first meeting will be held in Bratislava on 26 and 27 November 2007.

# Euratom et le rôle du nucléaire dans la politique énergétique européenne

*Dominique Ristori, Directeur général adjoint,*
*Direction générale de l'énergie et des transports*
*Commission européenne*

## Introduction

- En 2007 le traité Euratom célèbre ses 50 ans. A cette occasion il est intéressant d'analyser l'importance de ce traité dans le contexte actuel, de préciser la place du nucléaire dans la nouvelle politique énergétique européenne et d'identifier les actions prioritaires en cours.
- Plan d'intervention :

  I. Contexte économique, énergétique et climatique ;

  II. Grands axes de la politique européenne de l'énergie ;

  III. Rôle du nucléaire dans la politique européenne de l'énergie ;

  IV. Actions prioritaires dans le cadre d'Euratom.

## I. Contexte économique, énergétique et climatique

- D'abord le contexte car les événements énergétiques se succèdent à une vitesse très rapide et on ne saurait définir une politique énergétique européenne sans prendre en compte l'importance globale de ces questions.
- L'énergie demeure essentielle pour le bon fonctionnement de l'Europe. Or la période des approvisionnements bon marché pour l'Europe semble révolue. L'énergie risque de devenir un des obstacles majeurs au développement économique en Europe et dans le monde.
- Comme beaucoup d'autres pays dans le monde, tous les Etats membres de l'UE sont confrontés au triple défi du changement climatique, de la sécurité d'approvisionnement et de la compétitivité :

  1. Changement climatique

     La production et la consommation de l'énergie sont responsables de 80% de toutes les émissions de gaz à effet de serre dans l'UE. L'énergie est la source du changement climatique et de la majeure partie de la pollution atmosphérique.

     Jusqu'à présent ce lien très étroit entre l'énergie et le changement climatique a insuffisamment été pris en compte.

2. Sécurité d'approvisionnement

La sécurité d'approvisionnement est menacée sous plusieurs angles :

– 1er élément : une demande énergétique mondiale élevée et durable

Il y a cinq ans la plupart des experts n'avaient absolument pas intégré cet élément dans leur analyse. Or, c'est un élément fort et durable parce qu'il est porté pas des marches de très grande dimension, notamment le marché asiatique, qui est en premier lieu chinois et indien. Dans ces pays la consommation par tête en énergie est encore très faible et ne va cesser d'augmenter au fur et à mesure que l'urbanisation rapide se développe, créant de nouveaux appétits énergétiques liés aux nouveaux consommateurs des grands marchés asiatiques chinois et indien (voitures, articles électriques, etc.). Il s'agit donc bien d'un phénomène durable qui va influencer pour longtemps les marchés mondiaux, pétroliers et gaziers, mais finalement, tous les marchés énergétiques.

– 2ème élément, de nature mondiale aussi mais avec un effet européen direct : le manque d'investissement. Depuis deux décennies au moins nous avons vécu un peu partout dans le monde sur l'illusion de la surcapacité. Il n'y a pratiquement plus eu d'investissements. C'est le cas en Amérique du Nord, au Japon, c'est le cas en Europe, c'est le cas quasiment sur tous les continents, tant dans la production que dans le transport et la distribution de l'énergie. La conséquence est l'ampleur des investissements à faire et l'urgence des décisions à prendre.
Plus précisément, l'ampleur des investissements à réaliser pour le secteur énergétique a été évalue par l'AIE à 900 milliards d'euros sur toute la chaîne énergétique rien que pour l'Union européenne et à 20 mille milliards de dollars pour l'ensemble du monde d'ici 2030.
S'y ajoute l'urgence de prendre les décisions. L'âge moyen des centrales nucléaires dans l'UE avoisine les 25 à 30 ans.

– 3ème élément : l'accroissement de la dépendance externe. L'Europe dépend de plus en plus des importations d'hydrocarbures. Si rien ne change, la dépendance de l'UE à l'égard des importations d'énergie passera de 50% de la consommation d'énergie dans l'UE à 70% en 2030. Les crises en Ukraine et Biélorussie ont démontré les risques que peut entraîner une dépendance externe importante.

3. Compétitivité

En même temps en Europe, comme ailleurs dans le monde, les entreprises et les consommateurs subissent de plein fouet les tensions sur les prix de l'énergie. Ceci influencera les positions de nos dirigeants politiques tant nationaux, qu'européens.

L'énergie est un élément clé de compétitivité et, comme je l'ai indiqué en introduction, le manque d'énergie risque de constituer un des obstacles majeurs au développement économique.

## II. Grands axes de la politique européenne de l'énergie

Ce contexte général étant posé, voyons quelles actions politiques peuvent être entreprises au niveau européen.

D'abord la prise de conscience des Chef d'états et de gouvernements sur ces questions est une chance unique pour définir et mettre en place une politique énergétique européenne, qui utiliserait à plein la puissance collective que nous représentons en termes européens.

Lors du Conseil européen des 8 et 9 mars 2007 les Chefs d'états et de gouvernements ont fixé un objectif stratégique pour lutter contre le changement climatique : évoluer vers une économie à faible teneur en dioxyde de carbone ("*low carbon economy*"), limitant l'augmentation de la température moyenne mondiale à 2°C au maximum par rapport au niveau de l'époque préindustrielle.

En vue d'attendre cet objectif, le Conseil européen a prôné une approche intégrée de la politique en matière de protection du climat et de la politique énergétique, étant donné que la production et l'utilisation de l'énergie constituent les principales sources d'émissions de gaz à effet de serre.

- Protection du climat

  Au niveau de la protection du climat, le Conseil européen a pris l'engagement pour l'UE de réduire les émissions de gaz à effet de serre d'au moins 20% d'ici 2020 par rapport à 1990. En même temps le Conseil européen a appelé à des négociations internationales impliquant un objectif renforcé de l'UE à 30% de réduction des émissions de gaz à effet de serre, pour autant que d'autres pays développés fassent des efforts comparables et que les pays en développements, tels que la Chine et l'Inde, contribuent de manière adaptée à cet effort.

- Une nouvelle politique énergétique

  Sur base des propositions de la Commission, le Conseil européen a également adopté une sérié de mesures de politique énergétique visant à assurer la transition vers un système énergétique à faible teneur en carbone ("*low carbon energy system*").

  Parmi les mesures prioritaires, il convient de citer notamment :

  1. La transition vers un bouquet énergétique ("*energy mix*") basé sur des sources d'énergie faiblement émettrices de $CO_2$.

     Il s'agit d'une part de développer les énergies renouvelables, avec des objectifs contraignants de 20% d'énergie renouvelable dans la consommation énergétique de l'UE d'ci 2020 et de 10% de biocarburants dans la consommation totale d'énergie et de gazole destinés au transport d'ici 2020.

     D'autre part, il faut également tenir compte, pour les pays qui le souhaitent, de la contribution du nucléaire dans ce contexte, sur lequel je reviendrai plus tard.

2. L'accroissement de l'efficacité énergétique, avec l'objectif d'économiser 20% de la consommation énergétique de l'UE par rapport aux projections pour l'année 2020.

   Dans ce domaine les actions clés concernent les produits électriques (définition des normes d'efficacité énergétique), le secteur des transports (moteurs, carburants, gestion du trafic urbain) et les bâtiments (isolation) avec une proposition d'un nouvel accord international sur l'efficacité énergétique.

3. La diversification des sources d'énergie et d'approvisionnement en vue d'augmenter la sécurité d'approvisionnement, notamment en s'appuyant sur des mécanismes de solidarité et des actions de politique énergétique internationale.

4. La finalisation du marché intérieur de gaz et de l'électricité, en particulier en augmentant le degré de séparation entre les activités de production et de transport, et en élevant le niveau d'intégration des marchés (impliquant des investissements au niveau des interconnections transfrontaliers et de nouvelles tâches pour des régulateurs).

5. Un effort important dans le domaine de la recherche, sur base notamment d'un plan stratégique européen pour les technologies énergétiques, qui concerne principalement les énergies à faible teneur de dioxyde de carbone (sources renouvelables, le nucléaire et le charbon propre).

## III. Rôle du nucléaire dans la politique européenne de l'énergie.

Après avoir esquissé le contexte et les grands axes de la nouvelle politique énergétique européenne, je reviens à la place du nucléaire en Europe et à la contribution du nucléaire aux objectifs de politique énergétique.

- L'importance du secteur nucléaire en Europe

  Aujourd'hui prés d'un tiers de l'ensemble de la production électrique en Europe provient du nucléaire et l'UE est devenue le principal producteur d'électricité nucléaire dans le monde.

  Elle possède un parc de 150 réacteurs en activité, répartis dans 15 Etats membres.

  L'Union européenne a également le leadership technologique sur tout le cycle, y compris le traitement du combustible et l'enrichissement.

- La contribution du nucléaire aux objectifs de politique énergétique

  Dans sa Communication du 10 janvier 2007 sur "une politique de l'énergie pour l'Europe" (et le programme indicatif nucléaire, qui lui est annexé), la Commission a mis en exergue la contribution de l'énergie nucléaire aux objectifs de la politique énergétique européenne.

  Pour rappel, ces objectifs concernent les défis du changement climatique, la sécurité d'approvisionnement et la compétitivité.

  1. Changement climatique

     Premièrement, le nucléaire constitue une des principales sources d'énergie exemptes de dioxyde de carbone ($CO_2$) en Europe.

     Pour les Etats membres qui le souhaitent, le nucléaire pourrait ainsi faire partie d'un scenario énergétique, qui devra prévoir des réductions d'émissions importantes au cours des prochaines décennies.

  2. Sécurité d'approvisionnement

     Comme déjà indiqué, environ un tiers de l'électricité dans l'UE provient du nucléaire.

     Le nucléaire contribue ainsi à la diversification et à la sécurité d'approvisionnement à long terme pour les raisons suivantes :

     – la disponibilité suffisante et la répartition géopolitique des fournisseurs d'uranium dans des régions politiques stables (l'Australie et le Canada couvrent 45% des besoins de l'UE en uranium);

- l'influence limitée (10 à 15%) de la matière première (l'uranium naturel) dans le coût total de la production d'électricité;

- les capacités de production (large éventail de fournisseurs pour la fabrication et le transport; fournisseurs de l'UE pour l'enrichissement).

3. Compétitivité

Le nucléaire est une des sources à faible teneur en carbone les moins coûteuses actuellement disponibles dans l'Union européenne.

Les coûts du nucléaire sont relativement stables. En effet, comparé à la production d'électricité sur base du charbon ou du gaz, l'énergie nucléaire est moins sensible à la volatilité des prix du carburant, étant donné que l'uranium ne présente qu'une faible part du coût total de production d'électricité. En outre, comme déjà indiqué, cet uranium est suffisamment disponible et largement réparti dans le monde.

Rappelons également dans ce contexte que le nucléaire est un des secteurs où l'UE dispose d'un leadership technologique mondial (nouveaux réacteurs, enrichissement, retraitement).

- Reconnaissance politique

Il n'est pas neutre de rappeler que lors du Conseil européen des 8 et 9 mars 2007, les Chefs d'états et de gouvernements ont pris acte de cette analyse de la contribution apportée par l'énergie nucléaire aux objectifs de la politique énergétique européenne.

Dans ce cadre le Conseil européen a également souligné que chaque Etat membre est libre de choisir sa propre palette énergétique, et qu'il faut veiller à ce que la sûreté et la sécurité nucléaires occupent une place centrale dans le processus de décision.

Ceci nous ramène à la dernière partie de mon exposé, qui concerne les actions prioritaires engagées actuellement dans le cadre d'Euratom.

## IV. Actions prioritaires dans le cadre d'Euratom

En 2007 nous célébrons les 50 ans du traité Euratom.

- Importance du traité Euratom.

  Ce traité a permis d'assurer un niveau élevé de sûreté et de sécurité. Le cadre multilatéral conduit à la discipline collective et individuelle et donne les garanties que les travailleurs et les populations sont en droit d'attendre.

  Si les Etats n'étaient pas encadrés par les dispositions du traité, ce n'est pas la garantie mais l'incertitude, la disparité et le risque qui prévaudraient. Ceci a d'ailleurs été confirmé dès l'origine et a permis un fonctionnement adéquat des centrales nucléaires de l'UE depuis l'origine.

  Dans le cadre de l'élargissement, c'est la fermeté et la sévérité au titre de la sûreté qui ont prévalu pour inclure dans le traité d'accession le principe de la fermeture des unités ne répondant pas aux normes les plus élevées en Lituanie (Ignalina), en Slovaquie (Bohunice) et plus récemment en Bulgarie (Kozloduy).

  Ces normes sont indispensables pour l'ensemble des Etats et de la population, que ces Etats soient producteurs d'énergie nucléaire ou non, tant en termes de protection, que d'acceptation par l'opinion publique de l'énergie nucléaire.

  En outre, la protection sanitaire du traité Euratom a donné lieu à un corpus considérable de droit dérivé relatif à la protection de la santé des travailleurs et de la population.

  Le contrôle de sécurité des matières nucléaires dans l'UE poursuit deux objectifs principaux : assurer que les matières nucléaires ne sont pas détournées des usages auxquels elles sont destinées et garantir que la Communauté respecte ses obligations internationales concernant l'approvisionnement en matière nucléaire et l'utilisation de celles-ci.

- Actions prioritaires

  Dans sa Communication du 10 janvier 2007 sur une politique de l'énergie pour l'Europe, la Commission a réaffirmé le besoin de poursuivre, au niveau de l'Union européenne et conformément au droit communautaire, le développement du cadre le plus avancé pour l'énergie nucléaire, dans les Etats membres qui ont fait le choix du nucléaire, dans le respect des normes les plus élevées de sécurité, de sûreté et de non-prolifération, ainsi que le requiert le traité Euratom.

  1. Coopération renforcée avec l'AIEA

     Dans ce cadre la Commission a appelé à une coopération renforcée avec l'Agence internationale de l'énergie atomique (AIEA), tant au niveau de la promotion de la non-prolifération, que de la sécurité et de la sûreté nucléaires.

Ce renforcement de la coopération avec l'AIEA est déjà entamé depuis plusieurs mois et pourrait aboutir à la signature dans les prochains mois d'une nouvelle charte de coopération renforcée sur le triple créneau de la sûreté, de la sécurité et de la non-prolifération.

Un des dossiers particulièrement actuels et sensibles concerne la question du développement possible de mécanismes internationaux pour garantir l'approvisionnement en combustible nucléaire.

2. Sûreté et déchets nucléaires

L'Union européenne doit également poursuivre ses efforts pour inclure la gestion des déchets nucléaires et les questions de sûreté (y compris le démantèlement) dans ses futurs travaux. Il convient également d'assurer que ces normes exigeantes soient respectées au niveau international.

Afin de progresser dans ce domaine, la Commission va mettre en place, avec le soutien du Conseil européen et du Parlement européen, un Groupe européen de haut niveau sur la sûreté nucléaire et la gestion des déchets. Ce groupe sera chargé de développer progressivement une vision commune et, éventuellement, des règles européennes supplémentaires dans ces domaines.

En parallèle, et avec le même objectif à l'esprit de continuer à améliorer la sûreté nucléaire et la gestion des déchets, la Commission soutient les activités de recherche et de développement en matière de gestion des déchets. Ces actions sont conduites en particulier au titre du 7ème programme cadre européen de recherche et s'appuient sur une excellente coopération au sein de la Commission européenne, entre la Direction générale de la recherche, le Centre commun de recherche et la Direction générale de l'énergie et des transports. La plateforme technologique pour l'énergie nucléaire durable est prévue d'être lancée le 21 septembre prochain.

En ce qui concerne le démantèlement, la Commission a non seulement adressé une recommandation aux Etats membres, mais également renforcée sa coopération avec la Banque européenne de Reconstruction et de Développement (BERD), ce qui devrait être formalisée sous forme d'un "*Memorandum of Understanding*" devant être signé prochainement entre la Commission et la BERD.

3. Le Forum nucléaire européen

Une initiative récente de la Commission, qui a aussi été approuvée par le Conseil européen, concerne la création d'un forum nucléaire européen.

Ce Forum vise à améliorer les méthodes de gouvernance au travers d'une plus grande transparence et un dialogue renforcé entre la Commission européenne et les parties intéressées. Ce forum complète ainsi ceux déjà créés pour l'électricité (à Florence), le gaz (à Madrid), les énergies fossiles (à Berlin), les renouvelables et l'efficacité énergétique (à Amsterdam).

Ce Forum fournira une plateforme pour un dialogue structuré avec toutes les parties concernées –institutionnelles ou non – sur toutes les questions nucléaires pertinentes. Celles-ci pourraient par exemple concerner l'environnement industriel nucléaire, les moyens de rapprocher dans le domaine nucléaire l'industrie du

monde financier, le rôle des gouvernements notamment en ce qui concerne les normes de sûreté et de sécurité nucléaires, et les actions permettant d'assurer une information objective et fiable sur le nucléaire auprès du public.

Le Forum sera organisé alternativement par les gouvernements slovaque et tchèque, sur base de l'accord intervenu entre les deux Premier Ministres concernés et la Commission européenne. La première réunion du Forum est prévue à Bratislava les 26 et 27 novembre prochain.

## Conclusion

Nous vivons dans un monde en évolution très rapide, où les questions énergétiques prennent de plus en plus d'ampleur.

Pour les pays qui le souhaitent, le nucléaire peut apporter une contribution importante aux défis énergétiques actuels, notamment en termes de lutte contre le changement climatique, de la nécessité d'assurer la sécurité d'approvisionnement et de l'impératif de la compétitivité.

La volonté de la Commission est de poursuivre avec l'appui des Etats membres, des autres institutions et organisations internationales, toutes les actions permettant la mise en place d'un cadre légal le plus développé possible pour l'énergie nucléaire en matière de sûreté, de sécurité et de non prolifération.

Ce qui compte c'est de dégager une volonté politique forte en Europe pour atteindre ces objectifs. Il est également nécessaire de poursuivre cette politique en pleine transparence et de dégager - dans le cadre de débats ouverts et constructifs - les solutions les plus appropriées, en y associant toutes les parties intéressées, tant institutionnelles, qu'industrielles (aux niveaux production et consommation), la recherche, les syndicats, les milieux financiers et la société civile.

**Symposium ESARDA**

**Intervention de M. Olivier CARON**

**Gouverneur pour la France auprès de l'AIEA**

Mesdames, Messieurs

C'est en qualité de Directeur des relations internationales du Commissariat à l'énergie atomique (CEA) et de Gouverneur pour la France auprès de l'AIEA que j'ai le plaisir de m'exprimer devant vous aujourd'hui.

Je vous présenterai brièvement la manière dont sont appliqués les contrôles internationaux en France pour ensuite esquisser quelques pistes de réflexions sur les travaux d'Esarda.

Je sais que des présentations plus détaillées seront faites cet après-midi sur l'application des garanties en France mais je me permettrai toutefois de rappeler quelques faits significatifs sur les contrôles des matières nucléaires en France.

La France est un pays qui a une particularité. Il s'agit d'un Etat doté de l'arme nucléaire mais qui est aussi un des acteurs prépondérants dans le domaine nucléaire civil. Je rappellerai que plus de 78% de l'électricité en France est d'origine nucléaire et que nous maitrisons l'ensemble des activités du cycle du combustible.

L'importance de l'industrie française dans le domaine nucléaire tout comme l'engagement de la France dans la recherche nucléaire se traduisent nécessairement par un volume important de contrôles internationaux menés par la Commission Européenne et par l'Agence internationale de l'énergie atomique (AIEA). Ces contrôles viennent s'ajouter aux contrôles « nationaux » effectués par l'exploitant et par les services du Haut fonctionnaire de Défense du ministère de l'industrie (M. Sartorius, a d'ailleurs eu l'occasion d'ouvrir ce symposium et de vous présenter quelques éléments sur ce sujet).

La plus grande partie des contrôles internationaux sont effectués en France par la Commission Européenne qui doit être en mesure de contrôler l'ensemble des matières nucléaires civiles et peut accéder aux emplacements où elles se trouvent.
La France est ainsi, avec le Royaume-Uni, le pays le plus contrôlé au sein de l'Union Européenne.

L'AIEA exerce également aujourd'hui des contrôles sur certaines parties du cycle qui peuvent présenter une sensibilité particulière, notamment la réalisation de combustibles Mox. L'effort d'inspection de l'AIEA est cependant à ce jour assez limité en comparaison avec celui de la Commission Européenne.

Je profite d'ailleurs d'avoir autour de moi des représentants de l'AIEA et de la Commission Européenne pour réaffirmer que la France, mais je crois aussi une grande partie des représentants d'Etats de l'Union Européenne qui sont présents dans cette assemblée, reste très attachée à une bonne interaction entre les organismes internationaux de contrôle.

Mesdames, Messieurs

La mise en œuvre effective de contrôles internationaux en France est essentielle. Il est de notre devoir, même si les pays membres de l'Union Européenne sont irréprochables sur la non prolifération, de donner l'exemple et d'aider les organismes internationaux de contrôle dans l'exercice de leur mission.

Bien sûr des précautions doivent être prises pour ne pas compromettre l'efficacité commerciale de nos entreprises mais nous devons tous nous impliquer pour envoyer des signaux clairs à la Communauté internationale, à un moment où les crises de prolifération, nous le constatons tous, sollicitent nos efforts.

Dans ce contexte, Esarda a un rôle majeur à jouer. Tout d'abord, en étant un lieu unique de rencontre entre l'AIEA, la Commission Européenne et un grand nombre d'Etats Européens et parfois d'autres acteurs comme l'INMM ou certains observateurs. Cet atout doit bien évidemment être exploité pour faire progresser les réflexions sur le renforcement des garanties et la mise en œuvre de garanties intégrées. Esarda est aussi le cadre pour procéder à des retours d'expérience dans une démarche constante de progrès.

Cette association doit ensuite continuer à être un lieu de travail et d'échanges de nature technique, en particulier sur les équipements et technologies qui sont un des socles de l'application des contrôles internationaux.

Il va de soi que les défis en terme de détection d'activités nucléaires clandestines sont nombreux et que l'apport de vos travaux est essentiel. Les crises internationales

de prolifération nous rappellent chaque jour davantage que les contrôles doivent être adaptés et technologiquement innovants pour dissuader autant que faire se peut les proliférateurs potentiels.

La France entend participer et contribuer activement à ces réflexions. Je note d'ailleurs que les principaux acteurs français du nucléaire sont membres de cette association.

Je n'oublie pas non plus que la Commission Européenne joue un rôle très important au sein de cette association.

Mesdames, Messieurs,

La France entend poursuivre son action dans le domaine du renforcement des garanties. Elle est d'ailleurs impliquée dans les programmes de développement de nouvelles technologies pour renforcer les garanties et aider l'AIEA dans la détection d'activités nucléaires clandestines.

Je suis d'ailleurs heureux de rappeler que le programme français de soutien aux garanties de l'AIEA fête cette année ses 25 ans de fonctionnement. Grâce à une excellente collaboration avec l'Agence, ce sont aujourd'hui plus d'une vingtaine de tâches actives qui ont été recensées dans des domaines aussi diversifiés que les équipements, l'imagerie satellitaire, l'analyse, la formation, l'universalisation des garanties. La France met à la disposition de l'Agence une partie de son expertise et contribue avec un de ses laboratoires au réseau de laboratoires d'analyses agréés

par l'Agence. Outre des présentations sur ce programme qui seront faites tout au long de la semaine, je vous invite à vous rendre sur le stand français présentant des éléments de ce programme et qui a été organisé à l'occasion de la tenue de ce symposium en France.

Je tiens à remercier M. Heinonen, Directeur Général-adjoint de l'AIEA, responsable du département des garanties pour les propos élogieux qu'il vient de tenir sur notre programme de soutien et à l'assurer de notre volonté de poursuivre cette collaboration fructueuse.

Je conclurai en donnant quelques pistes de réflexion pour accompagner le développement d'Esarda de manière à ce que lors du prochain symposium en France (si les statistiques dont je dispose sont exactes) d'ici une dizaine d'années, nous puissions tous ensemble dresser un nouveau bilan positif des progrès réalisés dans le domaine de la non-prolifération et de la sécurité nucléaire.

Je crois que les défis technologiques peuvent être relevés notamment via un renforcement de la collaboration internationale. Esarda peut d'ailleurs jouer un rôle dans le renforcement de cette collaboration.

Esarda doit continuer à être un lieu d'échanges en particulier pour toute la communauté Européenne des garanties.

Esarda doit également poursuivre et renforcer sa politique de communication. Le monde doit savoir que les engagements sur ces sujets sensibles sont nombreux, cohérents et efficaces.

Ceux qui veulent tricher doivent savoir qu'il existe une véritable volonté internationale de réagir et de développer des outils de plus en plus performants et des concepts de plus en plus novateurs pour répondre aux défis à venir.

Esarda peut également jouer un rôle pour renouveler l'expertise dans le domaine des garanties. En effet, pour les garanties, comme pour tous les autres domaines liés au nucléaire, la préservation des connaissances, l'attractivité des fonctions est essentielle. Il faut savoir attirer des jeunes dans ces domaines pour relever tous les défis qui nous attendent.

Mesdames et Messieurs,

C'est à moi que revient le privilège de clôturer cette session et permettez-moi de vous adresser mes meilleurs vœux de réussite pour ce nouveau symposium. La qualité des orateurs et la richesse des programmes laissent augurer de fructueux débats.

De plus, le cadre enchanteur d'Aix-en-Provence n'est pas de nature à entacher votre moral, peut-être tout au plus votre assiduité. Mais le sérieux attaché nécessairement aux garanties viendra, j'en suis certain, contrecarrer cette tentation.

Mesdames, Messieurs, je vous remercie pour votre attention.

# Session 1

# Safeguards: historical perspective and future trends

# The EC - IAEA cooperation on implementation of international safeguards in EU

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**Abstract:**

*During the 2nd half of 2006, high level contacts between the EC and the IAEA resulted in a common commitment to further improve the cooperation in the field of the implementation of international nuclear safeguards in EU.*

*Technical meetings between representatives of the EC and the IAEA followed, identifying key issues for discussion such as the extension of the implementation of the Verification Agreement INFCIRC/193 to an enlarged EU, the implementation of common verification schemes using short-notice and unannounced inspections and the common use of equipment and information. The intention to progressively implement IAEA Integrated Safeguards in all EU countries was a focal point in the discussions.*

*The level of cooperation between the EC and the IAEA is based on a strong in-field sharing of experience and common execution of verification activities. The complementarity and the synergies of the EC and the IAEA are paramount for a credible, efficient and effective implementation of the two international safeguards regimes in EU.*

**Keywords:** INFCIRC/193, EURATOM Treaty, short-notice and unannounced inspections, common use of equipment and information

# Multilateral nuclear arrangements: Status and outlook

Bruno Pellaud, President of the Swiss Nuclear Forum[1]

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The concept of "multilateral nuclear arrangements" (MNA) has gained renewed attention as a way to strengthen the non-proliferation regime through the "de-nationalisation" of sensitive fuel cycle facilities in Non-nuclear Weapons States (NNWS). In early 2005, the IAEA Expert Group on MNAs reviewed the scope of such arrangements, from strengthened suppliers' assurances to the joint multinational construction of nuclear facilities. Since then, the IAEA has been the focus of serious discussions on a number of specific proposals for assurances of fuel supply to NNWS. Conversely, a number of countries have continued the development of their own national enrichment facilities, while others have expressed an interest in doing likewise, in particular uranium exporters eager to add value to their exports and to participate in the international supply market.

Assurance of supply for the operation of their nuclear power plants is the primary issue confronting NNWS, their first priority, ahead of general proliferation concerns. NNWS are unwilling to renounce their fundamental rights under the NPT (Non-Proliferation Treaty). Up to now, in spite of the small number of international enrichment suppliers – all connected to a NWS - the market of enrichment services has been fluid and mostly non-discriminatory. Nuclear plants operators in NNWS favour the continuation and the broadening of the present supply market.

Most MNA proposals come from supplying States: a voluntary commitment by a NNWS to forgo its own sensitive facilities would be exchanged against strong commitments to cover its fuel needs through supply-side commitments with or without IAEA involvement. What would be the incentives for a NNWS to enter into such arrangements? Economical, political? Should the internationalisation of sensitive nuclear facilities become the norm under the NPT?

The present paper deals with such questions from the perspective of small NNWS countries strongly dependent on nuclear energy for their economic and social welfare. How are the ambitious MNA schemes put forward in 2006-2007 in the name of non-proliferation to be reconciled with the deep concern of these countries to see their nuclear plants deprived of fuel through arbitrary and politically motivated supply restrictions? These “consumer countries” claim their own legitimate economic and political interests as fully “virtuous States” under the NPT. Economics is clearly in the foreground. Nonetheless, political considerations are not far behind, with genuine frustrations of seeing the nuclear weapons States unwilling to move an inch forward on the disarmament front (this year, refusing again to launch negotiations on a Fissile Material Cut-off Treaty in the frame of the Conference on Disarmament in Geneva).

## The MNA Expert Group

In mid-2004, the IAEA brought together a group of 25 experts from the technical and diplomatic communities - with the mandate to identify issues and options relevant to multilateral approaches for both front and back ends of the nuclear fuel cycle, and to provide an overview of the policy, legal, security, economic, institutional and technological incentives and disincentives for cooperation in multilateral arrangements. Two primary deciding factors dominated the assessment of multilateral nuclear approaches, namely "*Assurance of supply and services*" and "*Assurance of non-proliferation*". Both are recognised overall objectives for governments and for the NPT community. History has shown that it is quite difficult to find an optimum arrangement that will satisfy both objectives at the same time.

Whether for uranium enrichment, fuel reprocessing, or spent fuel disposal and storage, MNA options span the whole spectrum - from existing market mechanisms, up to co-ownership:

**Type I: Assurances of services not involving ownership of facilities:**

a) Suppliers provide additional assurances of supply
b) International consortia of governments provide additional assurances
c) IAEA-related arrangements strengthen assurances

The focus is here on reinforcing existing commercial market mechanisms on a case-by-case basis through long-term contracts and transparent suppliers' arrangements with government backing. *Examples: fuel leasing and fuel take-back offers, commercial offers to store and dispose of spent fuel, as well as commercial fuel banks*. Also, in developing and implementing international supply guarantees with IAEA participation: *Different models to be investigated, notably with the IAEA as guarantor of service supplies, e.g. as administrator of a fuel bank.*

**Type II: Conversion of existing national facilities to multinational facilities**

Concept: Promoting voluntary conversion of existing facilities to MNAs, and pursuing them as confidence-building measures, with the participation of NPT non-nuclear-weapon States and nuclear-weapon States, and non-NPT States.

**Type III: Construction of new joint facilities**

Objective: Creating, through voluntary agreements and contracts, multinational, and in particular regional MNAs for new facilities based on joint ownership, drawing rights or co-management for front-end and back-end nuclear facilities, such as uranium enrichment; fuel reprocessing; disposal and storage of spent fuel (and combinations thereof).

In its report of February 2005[2], the Group concluded that MNAs offer a potentially useful contribution to meeting prevailing concerns about assurances of supply and non-proliferation.

## Follow-up proposals

Since the publication of the IAEA Expert Group report, a number of proposals concerning *only* Type I have been tabled and extensively discussed.

### USA: Global Nuclear Energy Partnership (GNEP)

In February 2006, the US Government announced the launch of a "Global Nuclear Energy Partnership", an initiative[3] with very positive and far-reaching proposals, such as the expansion of nuclear power in the US, the development of advanced nuclear fuel cycles (including reprocessing), or the stockpile reduction of separated civilian plutonium. On the negative side, under the flag of non-proliferation, the GNEP would confine drastically the fuel service market. The USA, UK, France, Russia, China (and Japan?) would be "*Fuel Cycle Nations*" providing nuclear fuel to others (the "*Reactor Nations*") - in exchange for the commitment to forgo enrichment and reprocessing activities. A fuel-leasing plan envisages supplying enriched fuel for initial use in customer countries to be followed by its return, by chemical separation and by the burning of recycled materials in the "Fuel Cycle Nations". In a nutshell, the key elements of the civilian nuclear fuel (enrichment and reprocessing) would be fenced off and kept in the hands of Nuclear Weapons States running a kind of cartel. The man-made energy resource – plutonium – would flow to and be kept in the Fuel Cycle Nations and benefit only them.

GNEP is in essence a denial of technology based on the national policies and priorities of weapons States, but shrouded in well-meant non-proliferation principles. Some rare "reactor nations" – or more to the point, "consumer countries" – may well voluntarily consider such a proposal if attractive technologies and economic incentives are offered to them in exchange for renouncing national fuel cycle facilities. Time will tell – in a decade or so.

### Uranium enrichment industry

In May 2006, the World Nuclear Association (WNA) published a report[4] representing the views of a 28-member panel of nuclear industry experts regarding an industry-based backup supply mechanism.

The report tries to tackle the problem at hand through an unwieldy mixture of economical and political considerations that fail to take into account the interests of customer countries. Quite correctly, WNA states that any approach to strengthening security of supply should be consistent with the continued effective operation of the competitive world market and that any arrangement for emergency or backup or guarantee supply arrangements should be used only as a last resort when existing market arrangements have failed, and not as a substitute for market supplies. Unsurprisingly the uranium industry wants "*no price discrimination against supplies from the normal market, and hence no price subsidies for the emergency or backup or guarantee supply arrangements*". In truth, rather than receiving subsidies, the customers should – it seems - pay a premium for such

guaranteed supply arrangements! Unfortunately, WNA supports the political string that would be attached to such arrangements ("*To be eligible, a customer State must have made a commitment to forego the development of, or the building or operation of, enrichment facilities*"). Confronted with such a special "non-competition" clause, the customer countries will of course not pay a premium; they will not even consider such a restrictive arrangement without substantial economical incentives. Since the enrichers also want to be somehow compensated for the cost of providing such guarantees (e.g. dedication of inventory, construction of facilities, and actual supply costs), the proposed industrial arrangements will for sure require an ample source of third-party funding to satisfy financially both enrichers and customers.

**The Six-Country proposal**

In June 2006, six countries with commercial uranium enrichment activities – US, UK, France, Germany, Netherlands and Russian Federation, tabled a proposal[5] to offer 'reliable access' to nuclear fuel for States opting to rely on the international market for nuclear fuel and not to have domestic enrichment activities. Further conditions of admission are to be ascertained by the IAEA: to have a comprehensive safeguards agreement and an additional protocol in force, and to have no exceptional safeguards implementation issues outstanding with the Agency (in other words, the kind of countries having already a reliable access to the market…). The proposal refers to suppliers arrangements, to fuel reserves and to a limited broker role for the IAEA. There is no visible incentive for the customer countries, except the intent to consult them and a reassuring declaration of non-discrimination against the non-takers ("*Conditions of access to the commercial market for enriched uranium will not be affected for Recipient States that do not participate in this mechanism"*).

**Japan: standby assurance**

In September 2006, Japan proposed to establish a system[6] called the "*IAEA Standby Arrangements System for the Assurance of Nuclear Fuel Supply*" under IAEA auspices, that incorporates both an information system to contribute to the prevention of the occurrence of market failures and a back-up feature for supply assurance proposed in the Six-Country proposal[7]. This was a limited, but valuable proposal.

**UK: Enrichment bond**

In September 2006, the United Kingdom, in the context of the supply assurance envisaged in the Six-Country scheme, proposed[8] an "Enrichment Bond". This would enable "prior consent or de-flagging" for provision of enrichment services through the IAEA for qualifying recipient States. Germany and the Netherlands have associated themselves with this initiative. This means that the uranium suppliers will provide "*advance assurances that export approvals will be granted*" for further supply through the IAEA. Such a bond is of a major importance, since it lends the needed credibility to any IAEA-sponsored arrangement. Other suppliers of uranium or fuel services still need to express their readiness to accept such a scheme.

**IAEA fuel reserve**

Also in September 2006, the "Nuclear Threat Initiative" (NTI) – a US non-governmental organisation - proposed to set up a stockpile of low-enriched uranium under the Agency's auspices to serve as a last-resort fuel reserve for countries that have elected not to build a national uranium enrichment programme[9]. NTI offered a challenge grant of US$ 50 million

to be matched by US$ 100 million to be raised by the IAEA and its Member-States (in funds or in nuclear material). On May 23, 2007, the US House Foreign Affairs Committee approved a bill that authorises $50 million toward the same fuel bank initiative (as part of the $100 million matching amount). The IAEA is expected to develop the modalities of such a fuel reserve as to its technical and legal dimensions[10] (in particular as to qualifying criteria and release criteria).

**International enrichment centres**

Again in September 2006, Germany proposed the creation of an international uranium enrichment facility - operated by the IAEA at an extraterritorial (international) site[11]. The enrichment plant would be built as a "black box" and would only be accessed and maintained by the technology supplier. The plant would be built and operated on a purely commercial basis, without IAEA subsidies. This is a sensible but ambitious proposal. The advantages in terms of economics and non-proliferation are not evident, when this option is compared to the IAEA fuel bank concept fed from existing commercial nuclear facilities, since the release criteria would be practically identical.

Last October, Russia declassified the Angarsk enrichment facility and since then promotes it as International Centre under the IAEA[12,13]. There seems to be little difference with the EURODIF model that saw in the seventies a number of countries (Italy, Spain, Belgium and Iran) invest in the plant construction in exchange for assured fuel deliveries, but without access to the technology. The possible involvement of the IAEA is not convincing, since there is not much more "internationalisation" than under EURODIF. Nonetheless, Russia would welcome international investors to help expanding the facility. Will there be an ironclad assurance of delivery - in the light of Russia's inclination to flex its energy muscle occasionally?

Last, but still to be mentioned, Iran has suggested on various occasions the joint construction and operation of a multinational enrichment facility on Iranian soil. This is not likely to happen soon in view of the low level of international confidence in Iran's declarations. With the current obsolete centrifuge technology tested in Iran, nobody would *care*. With a more modern technology, nobody would *dare* for a very long time.

The academic world has also put forward some interesting ideas, including concepts that would combine assurances of supply with insurance and financing arrangements, a mix that could possibly provide economical incentives for customer countries to participate[14].

## Qualifying criteria – release criteria

Who would qualify to "benefit" as a customer country from the various proposals put forward? The common condition is that the country should renounce any plan to build a sensitive enrichment or reprocessing facility on its territory, not even – it seems - in a regional or international framework. This being apparently not enough, all schemes of assurances of supply start with the proviso: …"*provided the State is in good standing with the IAEA*", a short sentence that for some people should go as far as piling up all the desirable features: 1) a comprehensive safeguards agreement of course, 2) a ratified and implemented additional protocol, 3) good track records in nuclear safety and security, 4)

implementation of the United Nations Security Council Resolution 1540 on the non-proliferation of weapons of mass destruction and 5) solid nuclear export controls. For the most demanding pundits, the qualifying examination should include as well political/strategic considerations and the rationality of nuclear energy choices. *This goes too far; this is the best way to defeat the whole undertaking*. States fulfilling all these conditions will buy without restrictions whatever they want from the market and they will even build unhampered their own enrichment facility if they so wish on economic grounds. In order to attract less virtuous countries, the ladder should be set lower, as low as requiring only the first and third conditions here above. The same should apply to the release criteria to be adopted by an IAEA fuel bank or other international schemes.

## Waiting for customer countries

During the Special Event of the 2006 IAEA General Conference, a representative of the US government presented an impressive list of very good questions to address about each and all of the above proposals[15]. Still, he failed to ask the critical question, the essential issue to be investigated: "*How to make these proposal palatable to large and small non-nuclear weapons States in good standing – and even more important, to those that are not*?"

Without a doubt, finely chiselled schemes have been elaborated by nuclear industry, by fuel cycle States and by and for the IAEA under the heading "assurances of supply" – all apparently to the benefit of States ready to renounce domestic sensitive facilities. Where are the grateful guests? They are not yet at the door.

- States in good standing will not show interest for a long time in such complex arrangements of unproven value; they will stick to the open market (…and they will not be denied supplies). With a functioning market enlarged with the participation of additional suppliers, the customer countries will most likely eschew complicated fuel bank schemes managed by the IAEA; they will not give up the rights enshrined in the NPT.

- States not in good standing will not even come to the door unless *pulled* or *pushed* towards MNA schemes. Economical and other incentives (even possibly subsidised fuel) should indeed be devised to attract such consumer countries to MNA solutions. In the context of graded penalising measures following violations, measures decided by the Board, one can also imagine pushing non-compliant States towards MNA.

Nuclear weapons States have nothing to loose on assurances of supply with their large and closed fuel cycles. The NNWS, it's different: they must always learn at their own risk to balance cautiously their vital energy needs and their interest in non-proliferation. In the on-going MNA debate, the NNWS customer countries will listen, they should also make their own position heard more loudly, but they still need to be convinced on the need for sweeping MNAs.

In essence, the NNWS want a competitive nuclear market in which low-enriched uranium would be a "commodity" under stringent safeguards, but widely available from many

sources, and in which plutonium recycle would remain an open option for nuclear power operators. Is this view compatible with non-proliferation objectives? In principle, yes.

Nonetheless, with both objectives in mind – more nuclear power and stronger non-proliferation for the world – the nuclear community at large must imperatively readjust its plans and its vision in order to ensure a smooth development of nuclear power. The number of enrichment and reprocessing facilities cannot expand in proportion to the number of nuclear plants. Therefore, the likely scenario of a strong expansion of nuclear energy around the world calls for the development of nuclear fuel cycles with stronger multilateral arrangements – by region, by continent or by dedicated cooperation. Ancillary, but essential: Such multilateral facilities should not all be located in nuclear weapons States, so as to provide as much supply diversity as possible to those plant operators in non-nuclear weapon States with a vital dependence on nuclear power.

Schemes on assurance of supply and fuel banks (Type I) are unlikely to attract more than a handful of customer countries. The conversion of existing sensitive facilities into genuine international undertakings (Type II) meets with difficulties for the participating countries (obsolescence, national interest, security, safeguards implementation, financial and political risks). Of much greater importance are future facilities for which Type III schemes are the real solution. In NWS and in NNWS, multinational facilities should become the norm when a country, a region, a continent wants its own enrichment supply. South America, Japan and South Korea together, Australia and Canada together (as potential suppliers of enrichment services) and possibly small European countries together.

Such multilateral solutions have economical, commercial and political advantages. As such, they would not hamper the development of nuclear power, while undoubtedly strengthening the non-proliferation regime.

## A suite of ten considerations

As pointed out by Harald Müller[16] from the Peace Research Institute Frankfurt, all MNA schemes have been developed as “national supplier policies” without much consideration for consumer countries, giving an impression of discrimination between “haves” and “have-nots” bound to exacerbate antagonisms. Using the same terms, the former Director General of the IAEA, Hans Blix, noted recently in Berne that the NPT freezes the “haves” and “have-nots” in the possession of nuclear weapons; and that now the “virtuous countries” should accept to hand over the possession of the civilian fuel cycle to the same “haves”.

The basic question is indeed the one raised by Chaim Braun[17] from Stanford University: “Who is interested in implementing supply assurance proposals: the suppliers or the prospective users?” So far, only the supplier countries have spoken, while the consumer countries keep quiet, satisfied by the world market for fuel services and not yet ready to engage in restrictive practices without the offer of proper economical and political incentives.

Where do the customer countries – and in particular the customer utilities operating nuclear power plants - stand on the rich display of MNA proposals? Impossible to say yet; however, the following considerations may be worthwhile keeping in mind:

1. Small nuclear power plant operators located in small countries want a fluid and competitive market. Today, the commercial market satisfies the demand for fuel services; there is a diversity of commercial enrichment companies; enrichment capacity exceeds demand; and, based on current plans for the substitution of diffusion by centrifugation, capacity is likely to comfortably keep abreast of projected increases in demand in the medium term (e.g. until the end of the US/Russia agreement on HEU conversion to LEU). For other front end processes (such as conversion and fuel fabrication), the situation is similar.

2. The dependency on only a few enrichment suppliers located in and controlled by nuclear weapon States gives rise to concerns as to the continuity in the assurances of supply. Customer countries would welcome a greater diversity in fuel services and would welcome newcomers like Australia and Canada, countries that are already major players and reliable partners on the uranium scene.

3. Furthermore, to achieve an even more competitive fuel cycle market, the purchasers of nuclear fuel should seek a complete liberalisation of the market - with more suppliers - to achieve a perfect fluidity of supply. For example, this could be achieved through a "Commodisation of enriched uranium", the setting up of a kind of international "Chicago Commodity Market" for uranium dioxide, with twin entries: a low-value product at the natural 0.7% enrichment level and a high-value product at a maximum of 5%, each with a long-term market and a spot market. Physical mixing would provide the required enrichment just prior to fuel rod fabrication. Low-enriched uranium as a commodity can be easily stockpiled in a customer country or in a pool or cooperative of customer utilities.

4. As far as fuel element fabrication is concerned, there is no proliferation concern since fuel fabrication plants are not sensitive. Therefore, no need for ambitious international schemes. From a utility standpoint, as many such plants as possible in as many countries as possible makes sense. Group of countries or existing economic organisations (such as OECD) should see to it that a flexible and adequate fabrication capacity is always available for their own assurance of supply.

5. The overwhelming majority of consumer countries would probably be ready to renounce building *purely national* sensitive facilities, but not ready to give up the right to do so *multilaterally* with partners of their choice. For enrichment and reprocessing, they want to keep their commercial freedom to build such facilities jointly with like-minded (if it makes economic sense). Furthermore, they consider plutonium as a potential resource for their own good, with no inclination to leave that privilege to a few leading countries.

6. The issue of new multilateral facilities should indeed be addressed at the 2010 NPT Review Conference. Short of an impossible treaty amendment, as part of a broader bargain between NWS and NNWS, the Review Conference should decide that future

sensitive facilities are to be built in an appropriate multilateral or regional framework.

7. All other proposals (from the GNEP to international fuel centres) should be for a while kept out of the broader proliferation agenda and pursued separately on their own merits, as complementary measures to the open market. On the one hand, a number of consumer countries may well be interested in such incremental supply guarantees. On the other hand, supplier countries may be ready to offer attractive conditions to consumer countries willing to give up their own sensitive fuel cycle facilities. Let the two sides negotiate the proper terms in a series of bilateral arrangements, without attempting to create an all-encompassing framework under the NPT or in association with the IAEA. Experience over time will show those schemes most suitable for various partners, and, may be, some of them could later become worthwhile of consideration in the NPT context.

8. All the supplier-side proposals summarised above suffer from the same fundamental weakness, namely the lack of economic incentives. If the consumer countries are to come to the table, the promoters should do more than claiming noble non-proliferation aspirations. Substantial benefits in the form of price rebates or long-term economic clauses should be offered to the plant operators in order to entice their government to accept broader political constraints on industrial nuclear development for enrichment and reprocessing.

9. All IAEA related proposals (fuel banks and fuel centres) are confronted to diverging perceptions about the political independence of the IAEA. Over the years, the IAEA Board of Governors has functioned smoothly and efficiently when compared to other international bodies plagued by size and veto rights. Yet, the Board is eminently political, not always free from external pressures. An influential Board member – after having denied a fuel delivery – will do its utmost in the Board to prevent the IAEA to step on the scene as a substitute supplier. To give the IAEA a maximum of credibility for any of the proposals put forward, a clear distinction must be made between the role of the Board and the role of the Secretariat. It is up to the Board to write the appropriate guidelines and up to the Secretariat to implement them free from external interferences.

   Thus, a new major role for the IAEA requires two essential pre-conditions:

   - The delegation by the Board of Governors to the Secretariat of the operating competence for the implementation of "qualifying and release criteria" in relation to any fuel cycle activity of the Agency;

   - The granting to the IAEA of a generic “prior consent or ‘de-flagging’ by the suppliers contributing fuel to the IAEA facility, in other words, the recognition of the IAEA as end-user.

   In a word, consumer countries are unlikely to consider doing business with the IAEA, if the Secretariat and the Director General are seen or perceived exposed to the

double interference, that of the Board members and that of the suppliers delivering fuel or raw materials to the IAEA.

10. Before getting bogged down in the planning of administrative and technical details, it seems advisable to gauge the reasons for consumer countries to receive fuel from the IAEA, and to define incentives for them to do so. Dealing with the IAEA will imply a political risk for some (of having, so-to-speak, to "negotiate" with the 35 States represented on the Board, instead of one…), a risk that deserves compensation if an irreversible industrial and political commitment is expected from the consumer countries. How will these incentives depend on the non-proliferation credentials of the consumer country? Proportional or inversely proportional between a mere comprehensive safeguards agreement and a post-additional protocol "integrated clean bill of health". At any rate, in one form or another, incentives will be required.

## Concluding remarks

During the IAEA Special Event of September 2006, many NNWS expressed scepticism and concerns about the proposed MNA schemes.

The Minister of Minerals and Energy from South Africa, Ms. Buyelwa Sonjica, summarised[18] most eloquently the views of NNWS: "… *there is a need to guard against actions, which would merely serve to exacerbate existing inequalities, including through the creation of another kind of cartel that would exclude full participation, particularly by States in full compliance with their safeguards obligations … Although prevailing proliferation concerns may prompt us to consider alternative arrangements on supply mechanisms, these may under no circumstances impose unwarranted restrictions and controls over the legitimate peaceful use of nuclear energy … If we agree to such conditions, we may well be contributing to undermining the very bargains on which the NPT was founded and further disturb the delicate balance of rights and obligations under this instrument … In addition, we should guard against the notion that sensitive technologies are safe in the hands of some, but pose a risk when others have access to them".*

The chairman of the event, Charles Curtis, concluded the meeting with some sober observations[19]:

*"… establishing a fully-developed, multilateral framework that is equitable and accessible to all users of nuclear energy, acting in accordance with agreed nuclear non-proliferation norms, will be a complex endeavour that would likely require a progressively phased approach…Other unresolved key issues are how to structure assurance mechanisms in a manner that does not result in a real or perceived division between nuclear fuel/reactor technology haves and have-nots, and does not undermine existing multilateral, treaty-based nuclear non-proliferation norms or State sovereignty/rights".*

This is why it would be wiser to set aside the "*fully-developed multilateral framework*". This is not only very complex, but even impossible to achieve if treaty-based nuclear non-proliferation norms and State rights are to be respected. A gradual and loose strategy is the only way to go: firstly, by making attractive for the consumer countries all these various

proposals of assurances of supply, and secondly by focusing the broad political ambition to a single significant objective, namely, making multilateral arrangements the norm for all future sensitive nuclear facilities. On the first point, the sponsors should implement near-term projects with candidate countries to demonstrate their feasibility and attractiveness. On the second point, the non-proliferation community should work towards the 2010 Review Conference; if not possible there, it should seek a majority vote in the IAEA General Conference.

A recent (non)-paper from the European Union[20] has very correctly noted: "*As different States will have different motivations and interests, we should refrain from focusing on the idea of a uniform approach. A certain flexibility, taking into account the different national viewpoints, seems to be necessary. A step forward could be a mix of a limited number of multilateral mechanisms"*.

The IAEA is going in the same directions. On June 15, 2007, commenting on the submission to the Board of Governors of a yet unpublished report[21], the IAEA Director General stated, "*Trends clearly point to the need for developing a new multilateral framework for the nuclear fuel cycle. And it´s clear that an incremental approach, with multiple assurances in place, is the way to move forward … Such a framework is voluntary and States are free to choose their fuel options - no rights of States would be compromised".*

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# Fifty Years of Safeguards under the Euratom Treaty – A Regulatory Review

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***Abstract:***

*March 2007 marked the 50th anniversary of the signing of one of the founding treaties of the European Community. The Euratom Treaty has its origins at a time when the stability of energy supplies in Europe was a major concern. Recently, much debate has centred on the possible reform or repeal of some parts of the treaty, given that its original aim was to promote and oversee the development of nuclear energy in Europe. This debate has focused attention on the future contribution of nuclear power to increasing energy demands in an enlarged Europe. However, despite these issues there is near universal agreement that the Euratom Treaty has played a vital role in the protection of European citizens through the controls required for nuclear materials.*

*Chapter VII of the treaty (Safeguards) confers wide regulatory powers to the European Commission to ensure that civil nuclear materials are not diverted from their intended use as declared by the operators. This paper describes the early period of operation of the safeguards inspectorate, and gives statistics on the numbers and types of inspections carried out by the Euratom inspectors, and discusses from an operational point of view the value of inspection activities. Further, a critical appraisal of Articles 77-85 within Chapter VII is made. The paper also considers those safeguards requirements that are important to strengthen, in order to maintain a strong regulatory system to oversee future challenges, particularly in the context of increasing decommissioning activities within Europe.*

*It is noteworthy that fifty-years after the founding of the treaty, many of the concerns about security of energy supply have re-emerged. It is a measure of the vision and forward thinking of its founders that the treaty has successfully overseen the safe and secure development of nuclear power in Europe (which currently provides a third of its electricity needs) and despite the many changes and developments that have occurred, that the objectives concerning safeguarding nuclear materials have been met as intended. The controls envisaged at that time remain fully relevant today.*

**Keywords:** Euratom treaty; safeguards

## 1. Introduction

In the 1950s, nuclear power was heralded as a solution to future energy needs, and was poised for rapid expansion. Whilst technically capable of exploiting nuclear energy, Europe at that time lacked sufficient enriched uranium resources. The priority was for European community countries to rapidly develop the necessary technology and acquire nuclear material to successfully use nuclear power for their energy needs. As well as developing links with other countries for the supply of the material, there were research goals, sharing of information, and making best use of resources. To provide a cooperative means of sharing technology, to jointly develop the newly emerging nuclear power resource for civilian benefit, and to further European integration after the previous war, the European Atomic Energy Community (EURATOM) was established with the signing of the Euratom treaty in 1957 by the 6 founding member countries (France, Germany, Belgium, Italy, Netherlands, and Luxembourg).

The tasks entrusted to Euratom were many – to conduct research, to establish uniform safety standards for health protection of workers and the public, to guarantee the equitable supply of ores and nuclear fuels to users, to exercise the right of ownership of special fissile material, to facilitate commerce in the nuclear market, to establish relations with third countries and international organisations promoting civilian uses of nuclear power, and to ensure by appropriate supervision that nuclear materials were not misappropriated from declared uses [1]. From the outset it was recognised that to mitigate the risks of militarisation of the nuclear materials associated with the civil nuclear industry, a safeguards system capable of accounting for the movement and stocks of nuclear material was essential. Thus the dual role of the (Euratom) Commission was created – firstly to promote, but also to apply controls and regulate the holding and transfer of nuclear materials.

Today, there are many who argue that the Euratom treaty is obsolete, and that the original aims to promote nuclear power are out of step with current priorities. Some point to the democratic deficit in the treaty, the lack of accountability to the European parliament [2, 3]. Others criticise the dual regulator/promoter role of the European Commission. Much has also been written about the overlap of responsibilities with the NPT requirements of the IAEA and the functions of the two organisations.

In the 50 years since its inception, it is therefore pertinent to ask what has been the contribution of the Euratom treaty to the safe development of nuclear power in Europe. How well have the treaty objectives (Chapter VII) to control and safeguard the nuclear material been met? This paper concentrates on this latter aspect of the Euratom treaty objective, starting with very brief descriptions of the background to the treaty and then the key features of safeguards development, and statistics showing the growth in safeguarding activities, followed by an appraisal of the treaty outcomes.

## 2. Background to the founding of the Euratom Treaty

One of the primary ideas for a European Atomic Energy Community was to serve as a catalyst for the wider goal toward European integration through European Economic Community. The founders of the Community saw the potential of joint cooperation in the emerging nuclear power resource as an example of the benefits of community integration [3]. The period of the 1950's was also characterised by concerns about the limited sources of fuel oil, and the expanding energy demands of the post-war European countries. This was put into sharp focus by the 1956 Suez crisis that revealed Europe's fragile access to Middle Eastern oil reserves. At the time, individual countries in Europe had already begun to establish national nuclear research and development programmes, although much of the nuclear technology and nuclear material (enriched uranium) was in the hands of the USA, Canada and Great Britain. The "Atoms for Peace" initiative of the US in 1953 allowed the transfer of technology and materials to participating countries for civil nuclear power use under condition of strict safeguards to prevent diversion to military use. The original signatories to the treaty sought to accelerate progress by creating centres of knowledge and expertise as well as acquisition of the nuclear material for civilian uses.

However, the negotiations for the treaty were far from smooth. National interests continued to take precedence over community interests – for example in the desire to develop a national nuclear weapons capability whilst restricting the access of other countries to the materials necessary [4]. Divergent national interests, different economic and administrative approaches and the question of whether member states had the right to develop a nuclear deterrent meant that the final treaty was as much driven by political aims and concerns as the desire for economic gain from nuclear power. The treaty provisions reflect the priorities and conditions deemed necessary for the exploitation of nuclear power at that time. Under the treaty, the Euratom Commission (later the European Commission) acquired the status of a supranational regulatory authority for radiological protection, supply of nuclear fuel materials and nuclear safeguards.

The safeguards provisions reflected the US bilateral requirements, but gave Euratom direct responsibility for fulfilling security demands. Some aspects of the uniqueness of the safeguards arrangements are that they gave rights of inspection in all member states (including nuclear weapons states) through the provisions of Article 81 of the treaty – (inspection powers which are limited in the case of the IAEA). The defence clause of Article 84 exempts materials declared for military use from safeguards, and the Euratom treaty does not prohibit military use of materials by member states. Article 86 gives right of ownership of special fissile materials produced or imported to the Community.

## 3. Implementing Treaty Safeguards (Early Years)

A major task for the Commission following entry into force of the treaty was the enactment of legislation to define the safeguards requirements. In 1959 Euratom issued safeguards regulations (7 and 8). Regulation 7 specified the means for complying with Article 78 on declaration of operating characteristics of the installation for safeguards purposes, although initially debate centred on the application of this to defence establishments [5]. This issue was eventually settled in favour of the member state and gradually a uniform application of the rules was established. Regulation 8 defined the rules for accountancy, reporting of movements, material inventory and of inspection. Mid-1959 saw the start of monthly declarations of material movements by the facility operators. Initial visits to nuclear installations took place in the second half of 1959, and the first Euratom inspection took place at MOL in Belgium in April 1960. Regular inspections by nominated inspectors (initially a team of just 4 persons), followed from May 1960 as required by Regulation 8 [6].

As an indication of the type of facilities covered by the regulations at the end of 1959, it comprised: 49 active installations (9 research laboratories, 20 industrial facilities, and 20 mainly research reactors). Monthly figures on stocks and movement from these installations were being sent routinely to Euratom. By 1960 the Commission had gained sufficient experience that the USA accepted Euratom controls in such facilities as the sole control over nuclear material of American origin. Thus the Euratom safeguards system was established as the first regional as well as international operational safeguards system [5].

The growth in safeguarding activity in this early phase is shown in Figure 1 below, which shows the number of installations subject to Basic Technical Characteristics (BTC) declarations (regulation 7), the number subject to periodic reporting of material stocks and movements (regulation 8), and the number of inspections that took place.

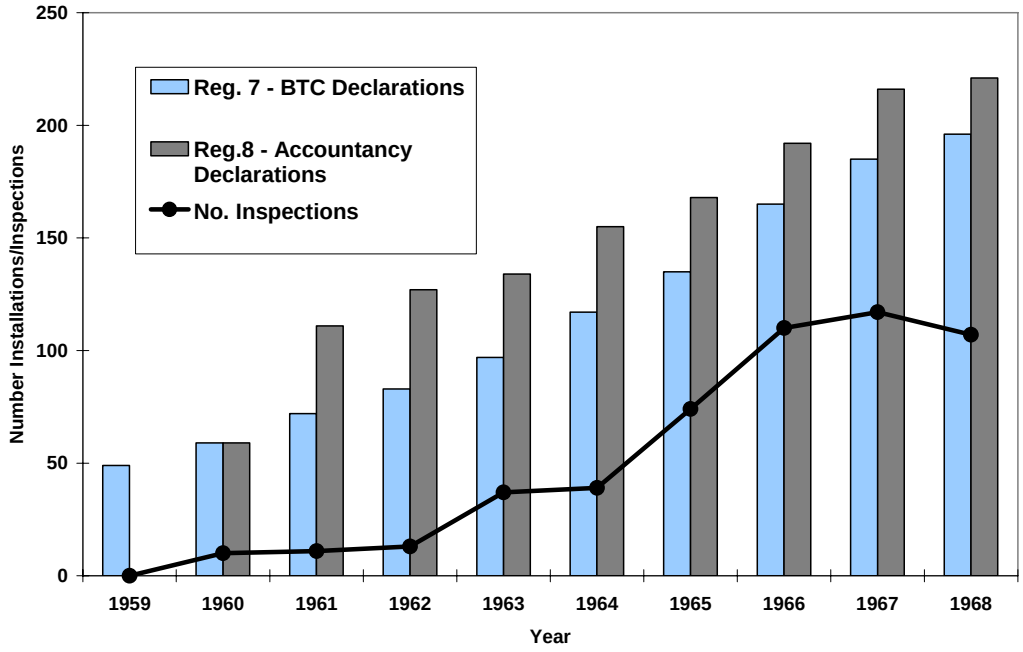


**Figure 1: No of Installations subject to Regulations 7 and 8, and inspection level (1959-1968*). (*Additional continuous inspections also took place in 1966-68).**

In 1962, Euratom began approval of the chemical processing techniques and plant characteristics for three spent fuel reprocessing plants. The first, the Eurochemic project at MOL, Belgium commenced operation four years later [7]. In 1963, the operation of the first full scale industrial power reactor (in

France) brought new challenges to safeguards. The expansion from research plants to full scale industrial plants called on new techniques to cope with verification of bulk raw materials and uranium hexafluoride gas rather than just finished fuel elements [8]. Safeguards verification in the early days was mainly based on accountancy declarations, simple mass/volume measurements or sample taking, but research was on-going to develop new instrumentation and measurement techniques. The inspection regime at the reprocessing plant called for continuous inspector presence initially, the control measures requiring US and Canadian authorities' acceptance for material of such origin [9]. The number of inspections in the period 1960-1967 by installation type is shown in table 1.

| Installation type | No of inspections |
|---|---|
| Fuel fabrication plants | 101 |
| Power reactors | 53 |
| Research reactors | 177 |
| Research centres | 50 |
| Irradiated fuel treatment | 20 |
| Fuel stores | 10 |
| Total | 411 |

**Table 1: Inspections by installation type 1960-1967**

The quantities of imported material under Euratom safeguards are shown below, illustrating the early dependency on imports of mainly enriched uranium. With the advent of the new power reactors from the mid-1960's the quantities of nuclear material under safeguards control started to rise.

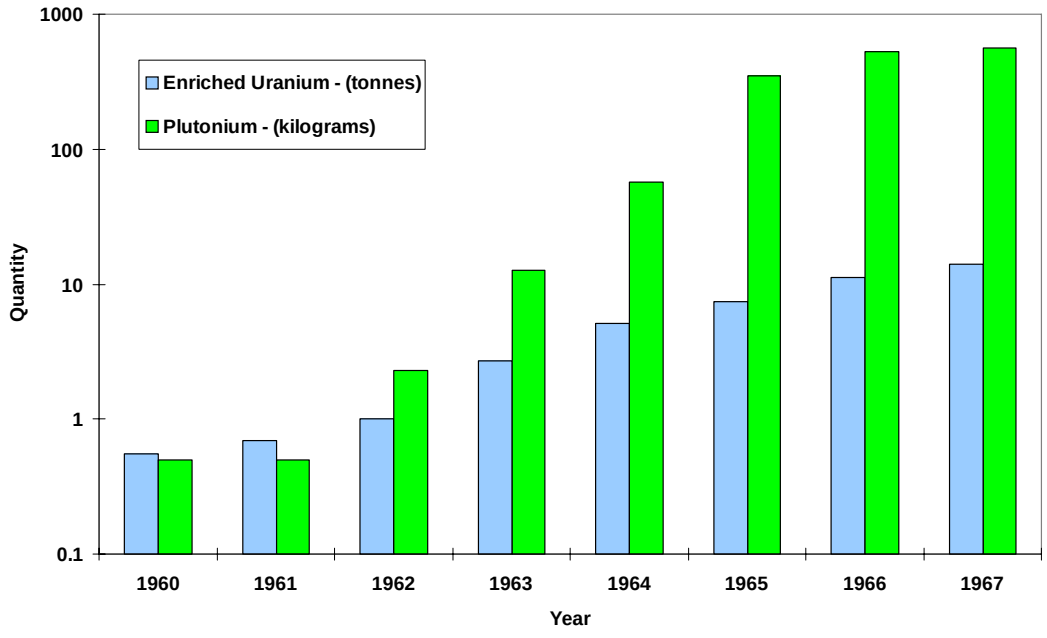


**Figure 2: Imported Quantities of Material under Euratom Safeguards, 1960-1967 (high- and low-enriched U, and Pu)**

## 4. Safeguards Development (Later Years)

The experience gained in these early years was of great importance for the future of Euratom safeguards. The late 1960's and early 1970's brought new challenges to Euratom treaty safeguards due to the negotiations for the Non-Proliferation Treaty (NPT). Euratom's regional safeguards system came under severe challenge and risked being superseded by overriding international non-

proliferation concerns. The desire to put global non-proliferation agreements in place put pressure on existing member states to accept IAEA safeguards in substitution for regional Euratom safeguards. Differences of view existed amongst member states, and further complications arose with the presence of the Nuclear Weapons States (NWS) initially France, and later the UK. Compromises had to be accepted that allowed both organisations to pursue their objectives in parallel. The INFCIRC 193 agreement defined the means by which IAEA would obtain independent verification of safeguards in the Non-Nuclear Weapons States (NNWS), whilst Euratom continued its regulatory role in the region. In the event, both NWS entered into voluntary agreements with the IAEA that allowed limited safeguards verification in their territory. The need for more formal agreements between the two organisations over the implementation of safeguards in the European community forced a redefinition of safeguards rules for accountancy, inventory change and material balance reporting. These were elaborated in the Community Regulation 3227/76, which was to remain the mainstay of Euratom safeguards regulation for the following 30 years.

In the early 1970's, nearly one third of the electricity production in Europe depended on oil [10]. The global oil crisis of 1973 drew attention to Europe's dependency on such limited resources. Nuclear power generation in the early 1970's began to show strong growth. The increase in nuclear facilities and the amounts of materials under safeguards can be demonstrated by the number of installations subject to safeguards and the quantities of material subject to Euratom control. Euratom responsibilities expanded further with the adhesion of key nuclear power countries, UK (1973), Spain (1985), and Austria, Sweden, Finland (1995). The effects of these events are described below.

## 4.1. Period 1969-1987

Table 2 below and Figure 3 show the rise in materials under safeguards control. The entry of the UK into the Community in 1973 resulted in a 50% increase in nuclear materials under safeguards control and a similar increase in inspection effort [11]. Further accessions in 1981 (Greece) and 1985 (Portugal, Spain) increased amounts under safeguards still further.

| Year | U-Nat | Enriched U | Pu |
|---|---|---|---|
| 1969 | 12500 tonnes | 16500 kg | 950 kg |
| 1970 | 13950 | 17146 | 1020 |
| 1971 | 13863 | 25461 | 1535 |
| 1972 | 15611 | 36635 | 1862 |

**Table 2: Quantities of Material under Euratom Safeguards, 1969-1972**

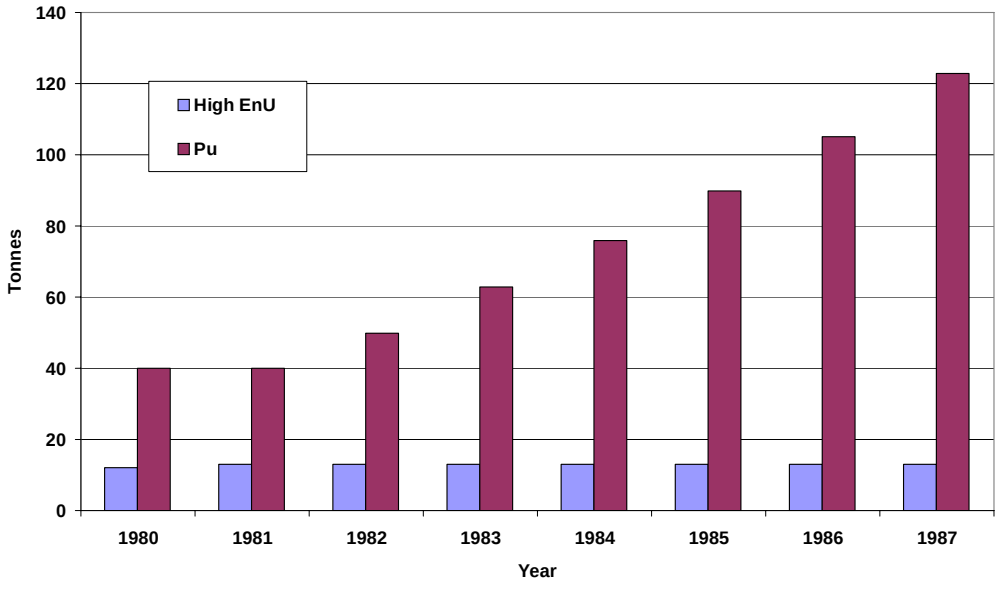


**Figure 3: Quantities of Material under Euratom Safeguards, 1980-1987**

## 4.2. Period 1988-2006

Safeguards controls developed still further throughout this period with joint cooperation agreements with the IAEA and new partnership approaches to rationalise still further the operations of the two organisations. Demand for nuclear power in Europe continued to rise, and this period saw inclusion of facilities in the new member states from 1995 (Austria, Finland, Sweden) under Euratom safeguards. The number of installations coming under safeguards control, as seen by the number of MBAs has continued to rise in this time, Figure 4. However inspection effort was dedicated to operations associated with higher risk. Currently, a major part (one third) of Euratom's inspection effort is dedicated to the reprocessing facilities at LaHague (France) and Sellafield (UK).

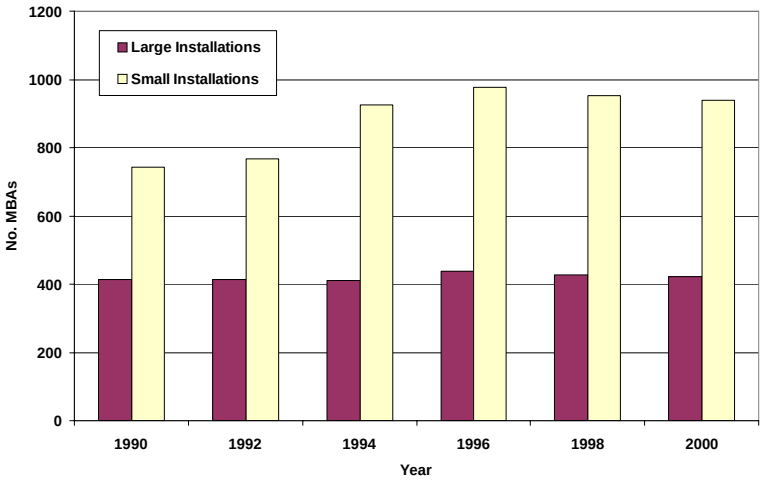


**Figure 4: Number of Material Balance Areas (MBAs) 1990-2000**

The equivalent quantity (as effective kg) under safeguards in the 10-year period from 1988 to 1998 increased by 188%, and in the ten year-period upto 2006, by 56%. This quantity is shown below in Figure 5.

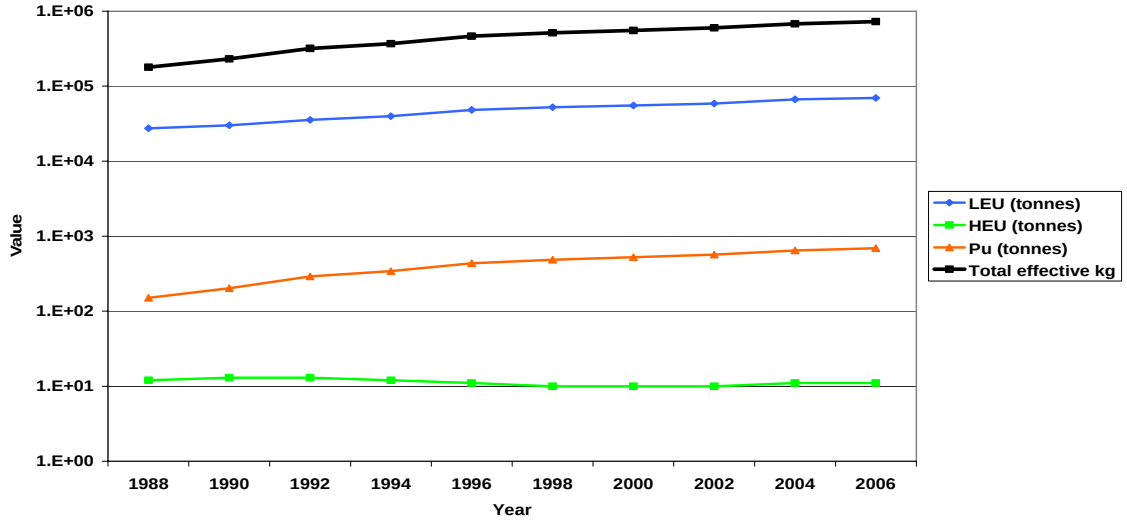


**Figure 5: Quantity of nuclear materials under Euratom-safeguards between 1988 and 2006**

The effect on inspection effort in the period to 1988-2006 is shown in Figure 6, demonstrating the effects of restructuring of Euratom inspection teams, and a policy toward reduced on-site inspection frequency.

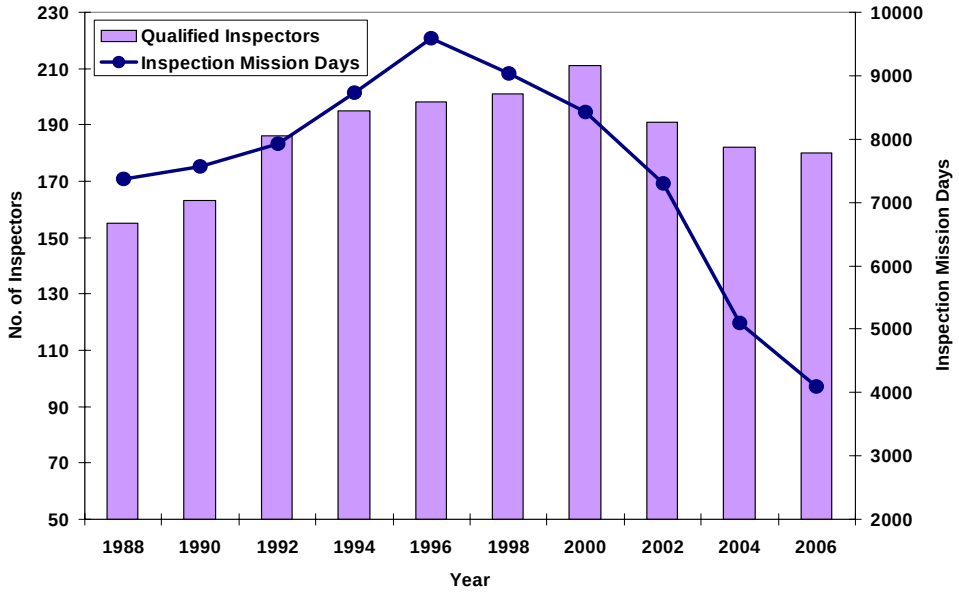


**Figure 6: Inspection Statistics, 1988-2006**

## 5. Review of Treaty Provisions

To meet safeguards objectives, the essential treaty requirements are stated simply in only 9 articles– (Articles 77-85) describing the essential features in a non-prescriptive, minimalist style. At its core are the basic functions to supply Basic Technical Characteristics (BTCs), provision of periodic operating and accountancy reports, and powers of on-site inspection. Although aspiring to community openness and transparency, member states were mindful of unwarranted intrusion in domestic and commercial affairs. It can be argued that a regulator should have greater rights to detailed information, and powers of scrutiny to fully assess the safeguards risks from the planning to execution stages of all nuclear projects. However, it can be said that this economy of regulation has been one of the reasons for the enduring nature of the regulations. The compromise treaty wording that was found to be politically acceptable at that time, remains in place today and serves its function.

The treaty confers wide enforcement powers ranging from issue of a formal warning, withdrawal of technical or financial benefits, placing the undertaking under administration or ultimately confiscation of the source materials. This ability to apply enforcement action on the operator or the member state is unique amongst the safeguards treaties. As a regulatory body, the Commission has not been in-active in using powers of sanctions under the treaty when required. To demonstrate the regulatory actions of the Commission there are examples of sanctions taken against both member states and operators of installations. Euratom has taken legal action against a member state (one case-Article 82), issued formal warnings to operators (seven cases-Article 83), or placed the undertaking under temporary administration (one case-Article 83) [12].

Regarding its adaptability to changing circumstances, the treaty does allow for alteration to the procedures for applying safeguards, under conditions of unanimous agreement of the Council. Herein lies the enduring nature of the treaty, in that with 27 member states the consensus for change would be far harder now. However the call for change lies with a minority of member states.

## 6. Appraisal of Euratom Safeguards

The formative period of Euratom was no doubt a challenging and rewarding time for those who worked in the organisation. The work took place in a new field that promised to yield many benefits through the civilian exploitation of nuclear power. It required a mix of disciplines, and in an era of fast change and constant growth, demanded strong skills in collaboration and cooperation amongst the regulators, researchers and policy makers. Within a few years of its creation, Euratom could claim to be operating a comprehensive safeguards system, which managed to provide reassurance to all member states, both nuclear and non-nuclear power states, that safeguards obligations were being met in the installations in the community by their operators. That represents the first such system to operate within a collection of nation states.

European safeguards needs have provided a strong driver for research and development that has contributed to the safeguards needs internationally. The Joint Research Centres have contributed for example to develop, test, calibrate and validate methodology, equipment and software for use by the inspectors, to train the staff in the technologies involved, and to support exploiting new technologies or approaches for both Euratom and IAEA. In the area of technical cooperation, Euratom and IAEA collaboration has been vital and yielded essential tools for common use. It can be said that the techniques developed through European research have application outside of nuclear controls. For example, it is thought that safeguards experience gained from control of civil nuclear materials can also be usefully applied to verification of nuclear weapons under the proposed Fissile Material Cut-Off Treaty [13].

The treaty provisions although developed at a time of 6 nation membership, has been adopted by 6 successive waves of accessions to the community, the most recent in 2007. Euratom successfully adapted to the demands of the nuclear power infrastructure of the new member states to the European community.

The value of inspection has been shown by their ability to detect discrepancies in operator records and declarations. These anomalies are subject to investigation and frequently are found to be due to isolated cases poor practice rather than systematic problems. In a very small number of cases operators have been found to be non-compliant and corrective actions requested. In some extreme cases legal action has been taken against the operator.

Controversy surrounds the benefits and successes of the treaty as a whole because of its origins as a pro-nuclear device. Many have criticised the Euratom treaty for the extent to which it has distorted the energy supply options in the last 50 years, and its relevance to current energy policy given the (uncertain) future of nuclear power. Regarding provision of economic aid for nuclear power, there is also some criticism of the lack of accountability to parliamentarians. All these are wider points worthy of debate, but it is worth noting that in the context of future energy policy, the same concerns about the stability of fossil fuel supply that existed in the fifties have remerged today.

Many argue that the commercial nuclear industry would have developed anyway – with or without the support of Euratom, however, the key feature of the treaty is that it made the development of nuclear power conditional on a strict system of safeguards. Most agree that concerning safeguards and the powers conferred by the treaty on the control of nuclear materials, the European Community has a good record and has played a vital role in the safe development of nuclear power. This achievement is not insignificant considering that the EU nuclear power industry has evolved to the point that it currently supplies 30% of its electricity needs. It is also one of the most highly developed commercial energy industries in the world, under strict regulations, providing a secure and reliable energy source that could not have been foreseen by the founders 50 years ago.

With regards to implementation of treaty safeguards provisions, some point to imperfections and possible lapses of control in the past. As is inevitable in the complex system of material handling and transport, there have been shortfalls in treaty compliance by operators and in the performance of the regulators. But it can be said that lessons have been learnt from these past lapses. The Euratom system of safeguards has provided reassurance to politicians, parliament and the public that strict controls do exist, operators are being carefully regulated, that obligations are being met. Given the political will and appropriate resources, much more could have, and can still be achieved here. Within a framework of regulation operators and member states recognise that Euratom safeguards serve an

important function – primarily to serve as an audit of their practices to pinpoint deficiencies, and when needed to enforce strict application of the rules [14].

As in the case of nuclear safety – it is agreed that for the effective and safe development of nuclear power it is essential to have an independent, highly effective and powerful regulatory authority to oversee its operations. The management of safety or security critical operations requires a strong regulatory authority with the necessary technical and financial resources to provide a high level service. In this context it has been shown that a strict system of safeguards not only assures material control for the purposes of non-diversion, but contributes to safety controls and safety performance, given the overlap of interest in maintaining a strict system of assurance and knowledge of processes and materials.

However, the main success of the treaty lies in the degree of community integration engendered by the safeguards arrangements. The ability of nationals of one country to verify implementation of safeguards in another neighbouring country by accord contributes to the transparency and confidence for establishing security in the region. These principles first enacted in the EC have resonance with the NPT non-proliferation aims and from the post-cold war era the nuclear arms control and disarmament phases in world politics. The experience gained in developing structures, methodology, technical skills and legal apparatus hold lessons for the other areas of arms control. It can be said that European safeguards control and monitoring – despite technical limitations, political interferences, complex relationships between members states, EU institutions, nuclear operators, and the IAEA - have achieved a major advance in international cooperation. The Euratom treaty can claim to have contributed to this achievement.

## 7. Strengthening Safeguards and Future Challenges

More recently the entire mission of the Euratom safeguards body has been questioned [15]. The non-proliferation remit and its selectivity (with reference to European weapons states) have been under scrutiny. In September 2000 a general discussion on the future of Euratom and its tasks was launched in relation to an internal reorganisation within the Commission framework. A High Level Experts Group (HLEG) was convened to make recommendations and in its report stated "….from a legal standpoint, Chapter VII…defines merely a nuclear material verification system under which accounting records, operating records and basic technical characteristics of facilities are properly kept by the facility operator and verified from time to time [by Euratom]." It is argued that excessive intrusion in operators' facilities is unnecessary since the non-proliferation aims are somewhat redundant in today's Europe, and that inspection regimes should be realigned to material security objectives. However, even this very critical overview of the safeguards function does not recommend a review of the treaty. The treaty remains relevant to current concerns – more so to do with security than non-proliferation.

It can be argued that the purpose of regulations is to confer some benefit, to provide clear rules about acceptability, and to describe a means for compliance, as well as operate as a deterrence against non-compliance. It is generally agreed that the Euratom system of control is well regarded by member states and operators. Current provisions are well accepted, well applied, and have provided confidence in the control of material in a period of rapid changes in the development of nuclear power. Concerning the issue that security of materials (against individual or group diversion) is the predominant risk, it could be argued that increased vigilance, and realignment of priorities is necessary rather than wholesale dismantling of treaty infrastructure.

However, as with all long established legal instruments, regular periodic review and redefinition of priorities is essential. It can be said that the Euratom safeguards authority (presently under DG-Energy and Transport) has been through a protracted period of introspection and scrutiny in recent years. What emerges is that the tasks of the organisation remain as important now as they were at any time in the last 50 years. Given that new threats exist today, it is of paramount importance that knowledge and expertise is maintained, that technical development continues, and that we do not become complacent to the inherent dangers in working with special nuclear materials.

In a climate of increased threats from loosely defined individuals and terrorist groups rather than through coordinated actions by nation states, the need for increased vigilance cannot be understated. To date, safeguards has only concerned itself with nuclear materials. However in the context of

concern about the possible misuse of other materials – attention should also be focused on safeguards measures for all high risk radioactive material. More so now than ever before, there is merit in redundancy of checks and verifications at every level.

The question remains, how to maintain a system of regulation which achieves the main objectives of independent verification, without being too complex, unwieldy, and burdensome on the operators? Much has been discussed in the scope of new approaches, improved efficiency and changes to safeguards provisions, eg to allow transfer of data, audit techniques, the need to incorporate new technologies, the use of more targeted inspections, the importance of separation of the operator's responsibility from that of the regulator (putting the safeguard obligation back onto the operator). However, what the treaty demonstrates is that safeguards demands do not require overly complex regulations. Over the next 50 years, the nuclear industry will be increasingly involved in decommissioning activities. These tasks, as well as the need to deal with legacy items bring many challenges to operators and regulators – requiring greater flexibility of approach but rigid demonstration of compliance.

## 8. Conclusion

The origins and development of Euratom's mandate were difficult and at times controversial particularly as it has been and continues to be a heavily politicised issue. It is therefore all the more remarkable that despite political and institutional difficulties in the last 50 years, the original treaty survives and its aims have been met. From very modest beginnings in the late 1950's, with a small core of staff and few facilities, European regional safeguards quickly established itself, and its expertise evolved to cope with one of the most advanced energy industries in the world. The figures show the rise in the quantities of nuclear material in use, representing the growth of the industry, and the large quantities under safeguards control today. The Euratom regional safeguards system continues to play an essential role in its regulation and control. For the demands and concerns of the European citizen, it can claim to be successfully serving its purpose.

However, decisions with respect to future contribution of nuclear energy are reaching an imperative stage. It is necessary to look at the treaty provisions critically and appraise the value of the regulations in relation not only to the future use of nuclear power, but on-going decommissioning liabilities which will extend to many years. It is clear that some treaty requirements could be amended or enhanced, for example shared decision making with parliamentary institutions would strengthen accountability and collective responsibility. Any amendment or translation of the treaty provision on safeguards should take account of forthcoming challenges. At a European level, the societal value of safeguards provided by Euratom should be reappraised, not least because the price to pay for even a single undetected real diversion would be beyond contemplation for the public.

The future development of nuclear power is a matter for the politicians and the public in each of the member states. For the service of the public, the supporting regulatory system must be able to provide an independent and trustworthy reassurance that safety and security aspects in the nuclear power industry are being treated with the importance they deserve.

*Any opinions expressed in this paper are those of the authors and do not represent the official point of view of the European Commission.*

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# Safeguards, between Technique and Politics

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***Abstract:***

*My involvement in international safeguards matters from 1968 until today form a treasure of historical facts, experiences, and memories. A resulting personal view and conviction emerged.*

*Nuclear safeguards is a tool to build confidence on the peaceful use of nuclear energy, based on hard facts and less quantifiable indications. The history of nuclear safeguards policymaking remained somewhere in the "no man's land" between science and technique at one side and the political decision making at the other side. The original credo "One World or None" used in 1945 for nuclear energy applies also today. Reflecting about the past might be helpful for building an improved future.*

*Once, the nearly universal acceptance of the non-proliferation of nuclear weapons was a forerunner of the global recognition of the technical reality, and assessment of the consequences of proliferation of the misuse of peaceful nuclear material and installations. Today quite differing approaches are followed to manage the global consequences of the proliferation of nuclear science. These political choices diverge from the ultimate logic necessities for supra national control of the peaceful application of nuclear energy, as they have been pursued (and sometimes rejected) in the past.*

*A review of the historical start, the development, and implementation experience with safeguards, the events and incidents that followed, and the solutions that are proposed today, all seem to confirm the fundamental concept that had been launched in 1946 for the first time.*

*The relation between technique and politics is one of love and hate, or trust and doubt. Placed in this field of tensions, nuclear safeguards play a role as mediator. It provides the basis of observable facts, and offers tools for rational weighing of less tangible indications for a righteous political development. That requires also a fair play between technical rationality and politically experienced feelings.*

## 1. Apology

In 2004 the idea arose to write a paper about safeguards between technique and politics. This idea was a result of observing the drastic changes, which at that time were going to be made in the Euratom safeguards. This came as a consequence of the strengthening of the IAEA safeguards during the previous years which resulted in the Additional Protocol.[1] A pyrrhic victory for the IAEA safeguards that got quite well specified restrictions. Euratom safeguards changed as a consequence more drastically than expected. My first reactions on that changes were in particular inspired by the report by the High Level Expert Group in 2002. That report laid the basis for the organisation and a new mandate for the newly formed Euratom Safeguards Office.[2] This office already replaced the previous Euratom Safeguards Directorate. But I was highly surprised to read that this office should enlarge its mission to the fields of safety, waste, and security. At the other side it should be constrained in the realm of its traditional safeguards, and was not allowed to acquire the newly developed technical possibilities, considered necessary for the

IAEA to go beyond the area of the declarations according to NPT Safeguards. Strange because the initial Euratom Treaty allowed its safeguards already to do all that is necessary. Hence a technical adaptation should be sufficient to create a body of Community competence in order to cope with the new IAEA rules, and settle eventual disputes that might arise between operators and IAEA about the interpretation of inspection results. At the other side the necessary freedom from the Euratom Treaty could not restrain Euratom safeguards by the formalities of the Additional Protocol. The political decision of the EC to change Euratom Safeguards was (and is still?) unpublished.

About the same time, the Convention for a new Constitution for the European Union decided to simplify the Euratom Treaty. In order to use less words, and make it more comprehensible for the citizens, essential articles defining the legal linchpin of the supra-national safeguards system were deleted. Luckily in the end the Euratom Treaty remained unchanged. However, the changes proposed by the High Level Expert Group, and the changes proposed by the political oriented representatives of the Convention, induced me to write a paper on the Euratom Treaty.[3]

By going to the roots of safeguards, it should perhaps be possible to get a better understanding of the backgrounds of the changes going to be made in the safeguards system of the European Union.

Due to personal circumstances I was however unable to come to the Esarda Symposium in London in 2005 with a follow up paper about safeguards between technique and politics for which an abstract had been submitted.

However the problem remained to intrigue me over the years, and I collected som me possible material. Hence last year the abstract above for this paper was submitted. Research in the developments in Science and politics in the first half of the 20-th century contributed to a better understanding of that thrilling period and the background of this revolutionary development of science. "One World or None" could describe the political climate at that time in the United States of America. It was also the Title of the popular edition of the Smyth Report[[4]]. It gives a comprehensive account of the nuclear weapons project, of the United States of America, ending with the question to the people of the US, to support a democratic decision on the question: how to proceed henceforth.

In the prevailing sentiments to start a new future with the United Nations and of "One World or None," lies also the origin of the Acheson-Lilienthal proposal. Now it became more intriguing to find out why the political presentation of the proposal for discussion on 14 June 1946, to the United Nations, by Bernard M. Baruch led to a predicted failure. Many political acts before that date had played their role.

To come to grip with that, search was continued in the history. The Truth is a difficult thing to grasp. It is like a diamond, from a distance it give a beautiful and brilliant impression. Going near to it there are too many facets to be seen all in detail from one position, and to see the whole.

Some subjects have crystallised out, but more research and writing is needed, to give a balanced view over this long period. Particular politics is an important factor, which was, and perhaps still is, a taboo in the safeguards world. But for a proper judgement of safeguards it has to be included. Finally my information research helped to find answers on old questions, to widen the personal view and strengthen my conviction. A secret can only remain a secret if only one person knows it. Honesty and openness in frank discussion with the other parties on the certainties and the limits of doubts is the only way to make a sustainable future.

On beforehand my apology for creating high expectations that have not yet been realised here, in particular for the history of safeguards policy making, as far as I have been involved.

## 2. Introduction

Changes in the world, e.g. internet, more publications on investigations in the past, the opening of old archives, and changes in political systems, made it possible to access a wealth of information that was kept secret for so long. A better understanding of the history is enabled, and it is now clear why the Acheson Lilienthal proposal was changed by Bernard Baruch and failed at the United Nations to start fruitful discussions for a United Nations Atomic Energy Commission.

Twice as much as is in this paper had been written, including several stories starting from the earliest days of uranium ore (dating back to 1512) onwards. Because that grossly sur-

passed the size of a readable conference paper, only some subjects, sometimes strongly abbreviated, shall be given as a prelude to more that I hope to present in the future. And other parts have been written specifically for this paper.

## 3. Understanding and doubt

### 3.1 Causality in science.

Werner Heisenberg realised that in the classical theory (applied to the electron movement) one could, on basis of abstract formulae, calculate the observables. Hence in 1925, he started along a similar lines from the classical theory, with abstract formulae to end up with the quantummechanical reinterpretation. He was fascinated by the idea, and considered later that at that time his work was nearly coercive. The first step of the work was finished about 12 June 1925. He showed it to two other scientists Born and Jordan and published it.[5] Those colleagues elaborated the mathematics of his model and soon found the first indication for the relation of indetermination, or uncertainty. They published "Zur Quantenmechanik I", and Niels Bohr followed with "Zur Quantenmechanik II". Therewith the new quantummechanics was born.

The uncertainty relations apply to the three pairs of canonical conjugate variables:

- position and linear momentum,
- time and energy
- the angle of the axis of rotation and the angular momentum.

At a festive dinner Heiseberg discussed the uncertainty relations and mentioned a fourth one, to the great surprise of the physicists present.

He exposed about a new pair of variables in which the increase in the knowledge of the one goes naturally at the expense of the clear knowledge of the other. This new pair was : deliberation and decision.

The philosophical consequences of the quantum mechanics, that for instance obliged to abandon the strict application of the causality principle, were subject of many discussions. The famous debate between Niels Bohr and Albert Einstein, started at the fifth Solvay Conference in Brussels. These regular conferences bring the worlds most notable scientists together. Electrons and Photons, the first particles that fit in the quantum mechanics initiated by Werner Heisenberg, was the subject of the meeting in 1927.

During the meeting Albert Einstein showed his disenchantment with Heisenbergs "Uncertainty Principle" by stating: "God does not play dice". Niels Bohr replied: "Einstein, stop telling God what to do". They continued their discussion as good friends and scientists by exchange of letters. The fundamental questions of Albert Einstein forced Bohr to improve insight in the consequences of the theory. But Einstein could not be convinced by Bohr's answers.

### 3.2 Living with uncertainty

The philosopher Karl Raimund Popper was fascinated by that discussion, studied quantummechanics, and concluded that also in daily life, we always will have to live with a certain amount of indetermination. Due to indetermination in the initial condition absolute correct prediction for the future is impossible. (A generalisation of the fourth uncertainty relation?)

In Karl Popper's work "The open Society and Its Enemies", short and perhaps too simplified given here, is argued that man have a free choice, are not bound to a fixed pattern. What we need in the open society are people with a critical rationality, that are able to express their views, and are open to enter into a debate. People do not belong to one closed group with a fixed pattern of behaviour, they live at the same time in different groups and exchange and develop their views. That should be the antidote to a totalitarian development towards a closed society that denies the fundamental individual freedoms[6].

In science, like in daily life, nothing is fixed forever. Reflection on the institutions is necessary from time to time in order to see whether they are still appropriate. In 1925 Heisenbergs quantum mechanics made clear that there is a margin of indetermination that cannot be removed completely. We have to live with this reality, in science and in daily life.

Confidence is nice, but it should not become blind faith. It must be possible to reconsider the eventual doubts from time to time on the most concrete facts.

Karl Popper vigorously defends liberal democracy and the principles of social criticism which he took to make the flourishing of the "open society" possible.

## 3.3 Politics and decisions

"I used to see politics as something bad. It's what caused our problems and made me a refugee for so long. But now I want to have a seat at the table." This statement was made by Athanasie Gahondogo, about her motivation to become a member of Rwanda's Parliament. [7] This is an example of accepting a responsibility for the political decisions. In fact in a democracy all the voters have their responsibility to communicate with their elected representatives about their opinion and wishes. In a non-democratic State at least the citizens indulge a government and have not all the desired opportunities to express themselves likewise. All states that signed the charter of the United Nations are in principle bound to the embedded rights for the citizens. But it is a weakness of the political solidarity between the citizens of the different states that not all states are coerced to respect those rights. Conflicts between aspects of the own security and human rights are often a decisive factor.

Here we come to the concise statement by the United Nations Secretary-General Kofi Annan in his report In Larger Freedom: "we will not enjoy development without security, we will not enjoy security without development, and we will not enjoy either without respect for human rights."[8] This statement opens an excellent approach to the creation and improvement of the general peace security in the world.

Politicians have to cope with conflicting norms, conflicting cultures, and conflicts between the personal and the common welfare and wealth. Two concepts that are not equal, in contrary they can differ exceptionally, from State to State, even when only the financial value counts.[9]

### 3.3.1 A sudden change

The power of the mass of people is difficult to understand. An extreme example is the change of government in 1933 in Germany, when the Nazi party came in power. All of a sudden hundred of thousands of people that were against the Nazi party, joined that party for a great variety of reasons. Perhaps an indication of the origins of such a mass movement can be found in the hierarchical distinction that Antonio Damasio made between emotions, feelings, memory of those feelings and the building of consciousness. As a neurologist using modern techniques, he is able to pinpoint the locations of the activities related to the different stages of this line of development inside the human brain.[10]

It may be speculative, but I found it elucidating in view of what I remember from all the documentaries about that period, which I have seen in the movies during my boyhood. These confronted us with the awful past in order to learn to live otherwise in the future.

Without having read Damasio's book, but learned from some critics and articles, it became clearer for me to see that the primary reactions in the form of emotions are strongly connected with survival which requests immediate bodily reaction, without any thinking at that moment. The consciousness resides at the end of the line from emotions to feelings and to memory. Consciousness needs rethinking to come to a reaction in which the personal rationality is expressed.

Not only the nuclear fission can release enormous amounts of energy. The history of the World Wars and thereafter has shown me that also the political leadership has the capability to unleash controlled or uncontrolled mass movements with incredible consequences, thanks to manipulation of the different levels from emotions to consciousness. But urgency diminishes the chance for a rational account of the actions taken.

### 3.3.2 Illustrative games

In 1979 the computer programmer Anatol Rapoport won with his program Tit-for-tat the contest for designing a computerised strategy for an optimal profit in a trade dilemma. The strategy worked fine in the simple conditions of the contest, but in real life more complexity should lead to an unmanageable system of accounting.

Another game, The Ultimate Game, allowed to reduce the problem of altruism against selfishness to a simple scheme that appeared to work in different groups of people from different cultures around the world. The rules of that play are: Player A has a sum of money

of which he can give something to player B. Does B accept, than the game can go on, does B not accept Player A has lost in this round, hence both don't win anything.

Economists guessed that B should accept any offer, but practice showed that the limits of acceptance all were near to about 40 percent. At an offer of 30 percent it was practically always refused. (a matter of human dignity?). The conclusion was that there are apparently natural limits on altruism and selfishness and a resistance to begging.

In the complex world of politics, day by day decisions have to be taken that have to be acceptable for all parties involved. It is interesting to see that in the described theoretical studies features, which for some people belong to common sense, now get some explicit scientific support. It shows possibly something of the rational and universal basis for human behaviour.

### 3.4 Esarda and uncertainties

Esarda working groups for Destructive Analysis and for Non-Destructive Analysis have a long history of defining standards, doing interlaboratory comparisons, and starting to collect the data that could quantify the uncertainties in measurement results. That belongs to the hard core of the concrete safeguards information that is available for verification.

Others, that are still active in this field can better report about the present, than I can do at this moment. The importance of the human factor on the measurement process changed with experience. And interesting new developments are to be mentioned. However today I'm unable to study in more detail this important subject for a proper presentation. However it might not be left unnoticed.

## 4. Acheson Lilienthal proposal

### 4.1 The group and its task

On 16 March 1946 a group of five man of quite different origins, had finished a comprehensive product of clear critical rationality and a convergence of their ideas in the report on the future of international control[11] of atomic energy[12]. When it was presented officially to the United Nations, it were the political circumstances and the pecular presentation of the report that led to a period of fruitless negotiations. Later the good concepts of this report should be integrated in the Euratom Treaty and the Statute of the IAEA.

The group was called together by the Commission on Atomic Energy of the United States, Secretary of State James F. Byrnes. That commission under the chairmanship of Dean Acheson, called those five man together in a Consulting Board[13]. Its chairman was David E. Lilienthal, Chairman of the Tennessee Valley Authority, an exceptional American utility, because it was not a company to make profit for the shareholders, but to provide a service to the people. One of its members was Dr. J. Robert Oppenheimer. He had resigned from his wartime office as head of the Los Alamos Laboratory, where the Atomic Bombs were made, and had returned to a scientific job at the California Institute of Technology and the University of California.

They started their work on 25 January 1946. The basis was a declaration by the President of the United States and the Prime Ministers of the United Kingdom and Canada, the "Agreed Declaration" of 15 November 1945. They represented the three States that, as far as they knew, had established a world monopoly on uranium, (and many believed also a monopoly on thorium and the technical knowledge of atomic power and weapons).[14]

This Declaration gives three fundamental points:

- The development of atomic energy, and the application of it in weapons of war, have placed at the disposal of mankind "Means of destruction hitherto unknown."

- There can be no adequate military defense against atomic weapons.

- These are weapons "In the employment of which no single nation can in fact have a monopoly."

The first General Assembly of the United Nations, held in London on 24 January 1946, adopted its first resolution for establishing the UN Atomic Energy Commission as a dependent body of the UN Security Council. Its ambitious programme covered the complete elimination of all weapons adaptable to mass destruction, but it was mainly concerned with “atomic” energy.

## 4.2 Working experiences

An important and essential aspect of the report lies in the way in which they had to work. Therefore the report also contains more words about this process.
"At the outset the vast difficulties of the problem were oppressive, and they thought at the start that only suggestions for various alternative proposals could be made without recommendations". They wrote in the report: "But as we steeped ourselves in the facts and caught a feeling of the nature of the problem, we became more hopeful. That hopefulness grew not out of any preconceived "solution", but out of a patient and time-consuming analysis and understanding of the facts that throw light on the numerous alternatives that we explored. Five men of widely differing backgrounds and experiences who were far apart at the outset found themselves, at the end of a months absorption in this problem not only in complete agreement that a plan could be devised but also in agreement on the essentials of a plan. We believe others may have a similar experience if a similar process is followed."

"There was no preconceived plan. It was the period of close study of the alternatives and an absorption in the salient and determining facts that led to this report."

Nearly all days from 25 January onward were spent on visits to plants and laboratories, to get acquinted with all facets of the problem, and many discussions.

## 4.3 The report

The development of the nuclear science in the first half of the 20-th century was the result of intensive co-operation between many scientists of many nationalities, and likewise the Manhattan Project was a result of such a broad co-operation, supported by nearly unlimited resources. Hence it is not surprising that this report has the refreshing character of a new way of thinking that is akin to the working in nuclear activities today, but that was less well understood in the society of 1946.

The report is presented as "not a final plan, but a place to begin, a foundation on which to build". It looks forward to the discussions to be held in the United Nations with: "Indeed, this process of joint international discussion is itself an integral part of any program for safeguards and security".

The report defines "dangerous activities", which could be used to make a nuclear weapon. That concept encompasses not only what is called today "sensitive technology" but also the mining of uranium. A feature that had been neglected in the NPT safeguards according to Infcirc/153, to the detriment of the completeness of the information, and the impossibility to close an overall balance[15]. Finally this loophole has been repaired by the Additional Protocol.

The dangerous activities should be placed under direct international control of the UN organisation. Because: "So long as intrinsically dangerous activities may be carried on by nations, rivalries are inevitable and fears are engendered that place so great a pressure upon a system of international enforcement by police methods that no degree of ingenuity or technical competence could possibly hope to cope with them."

The report is repeatedly stating that outlawing and policing are in the end counterproductive measures. International co-operation to stimulate the beneficial use of nuclear energy should be preferred.

It goes to far to describe the plan in all its details but a few other quotes may clarify what it is all about:
"It should be a plan that looks to the promise of mans future well-being as well as to his security."
"The beneficial possibilities in the use of atomic energy should be and can be made to aid in the development of a reasonably successful system of security."
"It is essential that a workable system of safeguards remove from individual nations or their citizens the legal right to engage in certain well-defined activities ...which ...will be generally agreed to be intrinsically dangerous because they are ... steps in the production of atomic bombs"

"But if the only legal ownership and development of uranium ore is in the hands of an international agency manned by and representing all nations, the problem of detection of evasions is, by a single stroke, reduced tremendously. ... The very opening of a mine by anyone other than the international agency is a "red light" without more ; it is not necessary to wait for evidence that the product of that mine is going to be misused."

### 4.4 Quoting: "Conclusions:

...[o]nly if the dangerous aspects of atomic energy are taken out of national hands and placed in international hands is there any reasonable prospect of devising safeguards against the use of atomic energy for bombs,

only if the international agency was engaged in development and operation could it possibly discharge adequately its functions as a safeguarder of the worlds future. Such a development function also seems essential in terms of attracting to the international agency the kind of scientists and technicians that this problem requires, recognizing that a mere policing, inspecting or suppressing function would neither attract nor hold them.

### 4.5 The moment to relinquish

The report only briefly discussed "stages" of implementation, but the moment on which the United States of America would clearly relinquish the bomb was considered to be handled separately.[16] This is another kind of negotiation as the more technical matters of setting up the Authority and its Charter.[17]

## 5. The other side

A few weeks before this Acheson-Lilienthal report was finished, on 5 March 1946 Winston Churchill had presented his lecture in Fulton at the Westminster College: "The Sinews of Peace". In this speech he coined the term "Iron Curtain" to describe the line in Europe between self-governing nations of the West and those in Eastern Europe under Soviet Communist control. [18]

But it was not generally understood at that time that in fact the horrors of Hiroshima and Nagasaki not only did mark the end of the second world war, the bomb explosions also were the start shot of the Cold War and the Nuclear Weapons Race.

### 5.1 Start of the Soviet programme

On 16 July 1945 at Alamogordo, New Mexico, Robert Oppenheimer and his team looked on in awe at the first man-made nuclear explosion. During the Potsdam summit meeting with Stalin and the British Prime Minister[19], Truman told Stalin on 24 July that the USA now had a new weapon of unusual destructive force. Stalin appeared unimpressed.

There are at least 7 eye-witnesses that have written about Stalin's reaction on this historic achievement. [20]

Remarkable are the observations of Winston Churchill, as described by him in one of his books. He had carefully watched Truman and Stalin speaking. He ends his story with: "... As we were waiting for our cars I found myself near Truman. "How did it go?" I asked. "He never asked a question," he replied. I was certain therefore that at that date Stalin had no special knowledge of the vast process of research upon which the United States and Britain had been engaged for so long... .[21]

In fact Stalin had learned about this test from intelligence communications already on 20 or 21 July. Stalin, however, did not expect that the atomic bomb would be deployed in the war against Japan so soon, in just two weeks.

At the Conference of allied powers in Yalta, in February 1945, the United States and Britain insisted that the Soviet Union join in the war with Japan approximately three months after the capitulation of Germany. The Soviet Unions agreement was contained in a secret protocol.

For Stalin entering the war with Japan was not only an act to help allies. He had serious strategic plans in Asia. The advance of the Soviet army into Manchuria, which was a colony of Japan from 1933, was only the beginning of their realisation. A communication about the explosion of the atomic bomb over Hiroshima on 6 August reached Moscow on the morning of 7 August. That day, at 16.30 hours, Stalin and Chief of the General Staff, A.E. Antonov, signed an order about the commencement of military actions against Japan along all the Manchurian border early in the morning of 9 August local time. That day, by order of Truman, the American airforce dropped a second atomic bomb, this time on Nagasaki. On 14 August, the Emperor of Japan announced the surrender over the radio.[22]

The reaction of Stalin has been reported as follows:
Stalin was furious and demanded quick actions:
Stalin was really enraged, that was the first time during the war that he lost control of himself... . What he perceived was the collapse of his dream of expansion of socialist revolution throughout all Europe, the dream that had seemed so real after the capitulation of Germany and was now invalidated by the

"carelessness" of our atomic scientists with Kurchatov at the top.[23]

### 5.2 The work and the workers

With surprising speed the Soviet nuclear weapon programme led to the first nuclear explosion of a plutonium bomb on 29 August 1949 in Semipalatinsk in Kazachstan. That programme had the advantage of some 300 tons of uranium that were taken from Germany at the end of the war, and a massive body of sensitive information obtained from the Anglo-American programme by espionage. Also the vast number of people working voluntarily or coerced should be mentioned as well as the incident in January 1949 when an major radioactive contamination occurred in the closed "Atomgrad", in the plant where the plutonium for the first bomb was separated. The many people that worked there had been forced to stay and continue the working in this plant in Tomsk. Among those people was the chief scientist I. V. Kurchatov. Also he fell ill for the rest of his life. It was about 1951 that I read for the first time about this incident without date and only the mysterious location Atomgrad. During the research for this paper I discovered that the story described cruel reality.

## 6. The Baruch plan

For the presentation of the Acheson-Lilienthal proposal to the United Nations, President Truman elected Bernard Baruch, an old and experienced diplomat. But when the writers of the plan got informed about that choice, they immediately realised that the plan would not be accepted. However they, and many other people, were unaware of the fact that the Cold War and the Nuclear Weapons Race already were pursued in full rigour.

Baruch's presentation speech stressed the compliance by strong measures. Among other things he said:

Penalization is essential if peace is to be more than a feverish interlude between wars. And, too, the United Nations can prescribe individual responsibility and punishment on the principles applied at Nuremberg by the Union of Soviet Socialist Republics, the United Kingdom, France and the United States - a formula certain to benefit the world's future[24].

"If I read the signs aright, the peoples want a program not composed merely of pious thoughts but of enforceable sanctions - an international law with teeth in it. "

When an adequate system for control of atomic energy, including the renunciation of the bomb as a weapon, has been agreed upon and put into effective operation and condign punishments set up for violations of the rules of control which are to be stigmatized as international crimes, we propose that:

1. Manufacture of atomic bombs shall stop;
2. Existing bombs shall be disposed of pursuant to the terms of the treaty; and
3. The Authority shall be in possession of full information as to the know-how for the production of atomic energy.

The process of prevention and penalization - a problem of profound statecraft - is, as I read it, implicit in the Moscow statement, signed by the Union of Soviet Socialist Republics, the United States and the United Kingdom a few months ago. 26 dec 1945.
*I feel obliged to note here that this is a particular interpretation of Bernard Baruch because careful reading of that statement about the commission that should deal with atomic energy shows that even nothing points in the direction of penalization. The most near to this point comes the text reproduced in ref.[25]. This declaration should become also the first resolution that the United Nations adopted on the atomic energy commission in London on 24 January 1946. And this comment is logically consistent with the Soviet signature on that declaration.*

After having listed the punishable offences Baruch continues with:
It would be a deception, to which I am unwilling to lend myself, were I not to say to you and to our peoples that the matter of punishment lies at the very heart of our present security system. It might as well be admitted, here and now, that the subject goes straight to the veto power contained in the Charter of the United Nations so far as it relates to the field of atomic energy. The Charter permits penalization only by concurrence of each of the five great powers - the Union of Soviet Socialist Republics, the United Kingdom, China, France, and the United States.

I want to make very plain that I am concerned here with the veto power only as it affects this particular problem. There must be no veto to protect those who violate their solemn agreements not to develop or use atomic energy for destructive purposes.

The bomb does not wait upon debate. To delay may be to die. The time between violation

and preventive action or punishment would be all too short for extended discussion as to the course to be followed.

*end of quotations*

It has to be remarked in relation to the veto power that the Soviet Union in the General Assembly of the United Nations could count on only three votes, from Russia, and two other states of the Soviet Union: Ukraine and Belarus. All the other states were initially no supporters of the Soviets. Hence the veto power in the Security Council was considered as a political necessity by the Soviets. Until today the veto remains a point of contestation that often blocks progress.

The Soviets, of course, could not agree with an obligation to accept inspections on its nascent nuclear weapons programme before the United States should have abolished, and dismantled its nuclear weapons and allowed inspections to verify this.

The negotiations that the Acheson Lilienthal plan hoped for, turned out to be impossible as a consequence of the higher politics, and the presentation at the United Nations of Baruch's own Plan finally closed the door for any kind of United Nations Authority for nuclear energy.

## 7. The Acheson Lilienthal heritage

### 7.1 What happened

Overlooking the history of more than sixty years, the world is still confronted with the same type of stalemate. The nuclear weapons have been used in many instances to deter successfully. Luckily these weapons had yet not to be used physically, but in some cases they were brought nearer to their target (Suez crisis, Cuba crisis etc). Today cases like India, Iran, Israel, North Korea, and Pakistan remain in the focus of public attention and political concern. Closely connected to the nuclear proliferation considerations, are the security problems of these states and of their regimes. That is the point the politicians have to solve.

Safeguards can be of help only when it is applied fully fledged, with an appropriate mandate, personal, equipment, funding, and political support. But often safeguards is in the political process an afterthought. The parsimony of all states towards the safeguards cannot be explained by lack of technical knowledge, or understanding. Perhaps there is a dangerous political gamble going on, as consequence of a desire to create room for manoeuvring by not giving fair technical insight to reduce the uncertainties about perceived intentions. Safeguards is an instrument of primordial importance. Should the named states have accepted a good form of international safeguards, a lot of suspicions and accusations could have been removed from the negotiation table. Illustrative is the Iranian approach that seems to use the level of accepted safeguards as a negotiation chip.

### 7.2 A cultural change

It is ill fate that the Acheson - Lilienthal proposal, as a technical and managerial answer to the first official decision of the United Nations about an Atomic Energy Commission, became a victim of power politics. The concepts of that plan have obtained much historical proof of their, nearly physical, correctness by the piece meal approaches discussed over the years without getting into realisation due to a lack of cohesion in a comprehensive political, technical and commercial whole[26]. The eldest example of a loyal and comprehensive implementation of the principles can be found in the Euratom Treaty embedded in the Community Treaties, and combined with a wide scope of other treaties, aiming at human rights and security.[27]

Perhaps the world is in small steps gaining the insight that we have to grow in that, once formulated, solution for nuclear energy and other large scale challenges for the modern world-society. It needs a change in culture. Power politics aims at sticks instead of carrots. The consequence is fear for punishment, which is counterproductive. As applied in the factory, e.g. at the basis, the workshop, it leads to hiding mistakes. That is going to change. In modern manufacturing the employee is getting back his personal responsibility to the benefit of the quality and continuity of the production and an increase in the social virtues, as described by Francis Fukuyama in his book Trust[28]. In the operation of nuclear plants the safety culture is another name, and claimless declaration in the hospitals is another translation of this general cultural change, which is imperative for coping with the vulnerabilities of the modern technically developed societies.

Also in the political field a balance has to be found between the power politics needed to defend the national interests and the broader and more idealistic oriented politics needed to serve the global commonwealth. Referring

back to the computer games of section 3.3.2, like has been indicated in the simplified situation of "The ultimate game", and discussed in the book "Trust", prosperity flourish on a basis of mutual confidence and responsibility. That includes a recurrent feeling of uncertainty, and as Werner Heisenberg argued (see section 3.1) that can be minimized on one aspect, but tends to increase in another complementary aspect. A balanced interaction on different sets of complementary variables can help to decrease and assess the remaining doubts.

## 7.3 European Union

Where a global approach to the problem of a United Nations atomic energy authority failed, another revival of the concepts came in an international global setting with the Statute of the IAEA and the regional application in the Euratom Treaty. The IAEA has been bound by the great differences between the member states. The founding members of the European Communities have common interests and showed a gradual merging of their different interests, originally in different Treaties. Recently it has been tried to merge them into one constitution. The European approach has great possibilities. But by the short sighted focus on the direct financial contributions paid and subsidies received, the political leaders tend to reduce cost at the expense of creating a good atmosphere of loyal co-operation for common welfare and wealth.

### 7.3.1 Committology

The European Union has two unique concepts. The Brussels jargon for these (complementary) concepts are committology and subsidiarity.

Mark Rhinard investigated the operation and the democratic legitimacy of the European Union committee system[29]. Committees make a critical contribution to the effectiveness of the EU policy-making system. Committees are a highly suitable institutional mechanism for consensual and depoliticized decision-making: small, insulated groups allow for extensive, ongoing consultation in an intimate atmosphere among actors from multiple levels of government and society.

Around 1968 already about 1500 meetings per year were convened which involved some 16 000 civil servants and experts from the member states. Despite their contribution to the success of the “European project”, committees are increasingly coming under attack, notably for their lack of democratic credentials. And that is understandable due to the decrease of scientific independence and the increased influence of commercial and political interests. Mark Rhinard proposed in the quoted article some measures for improvement.

### 7.3.2 The Ascent of Europe

Mark Leonard wrote a book titled: The Ascent of Europe.[30] He opens the quoted article with: "For all the talk of American empire, the last two years have demonstrated the limits of US power." America’s economic lead over the rest of the world has disappeared. Like others the difference lies in "hard" and "soft" power. The war on terror is provoking instability and fears, and Europe sees a world where everybody is a potential friend. The recent history obliges us to change the way we think about power. The journalists tell many stories about European Union's crises, however historians claim that it becomes increasingly stronger after each crises. That is the interesting point made in the article.

Remarkable, like Mark Rhinard, also Mark Leonard sees that: Many people complain about European red tape, but paradoxically it is the size of Europe’s body of laws that allows its institutions to be small. Further discussion of the differences between the United States of America's model and the European Union model concludes the success of the last one: Europe’s success has also set off a regional domino effect that could change the nature of power beyond its borders. In every corner of the world, countries are drawing inspiration from the European model and nurturing their own neighbourhood clubs from Asean and Mercosur to the African Union and the Arab League.

### 7.3.3 How to proceed

In 2004 the Dutch Minister of Foreign Affairs Bernard Bot, gave a lecture at the University of Oxford entitled "Great game or European fair play?"[31] In the speech he refers to Isaiah Berlin, who concluded that “there is not a fixed, and yet there is a common human nature: without the latter there would be no possibility of talking about human beings”. In the lecture he ties this to the European Union motto “Unity in diversity”, and claims that the Union is therewith in a strong position to foster the forces that bind rather than divide us.

Bernard Bot elaborates on the questions: Can the politics of principle eclipse the politics of power? Or is Europe setting itself up for a fall, showing naive optimism in a world dominated by zero-sum thinking? By going through the relations between the Union and the world, he comes to conclusions.
For the European Union, and the world, there is no acceptable alternative to fair play. And in this globalised world, there is a new rule: what is not good for you cannot truly be good for us. After giving seven conditions for the fair play in the Union's external relations, he concludes with Isaiah Berlin wrote: “[A]ll human beings must have some common values or they cease to be human, and also some different values, else they cease to differ.”

## 8. Concluding this paper

Looking to the past is like looking to a great stage where different plays are going on at the same time. Now a lot of secrets of the past have been published, and the plots with their many interactions and relations in time and place can be compared in the context of the past and the future for that moment and their interactions.

The Acheson Lilienthal plan stresses the process which they followed to find a logical and unanimously accepted solution that gave hope for the future, while at the onset the divergence of views and personalities, and the problem itself were of a depressive dimension. It assumed a fair play, but history developed with great uncertainties and consequent risks that nobody could accept. Some of the facts that are now unveiled, show that there was a lack of openness and fair play. A curious example in the American Russian relation is given by the execution of the transaction in the period 1942 to 1944, by which the Americans extracted uranium in Kyrgyzstan from local ore. By that action the Russians learned from the American plant how to treat the carnotite ores. But I should stop. Perhaps I can tell or write that story another time.

Belief in the power of weapons, or in punishment doesn't lead to peace. but only to destruction.

The history of safeguards has been written by cases that could have escalated in greater mistrust, but thanks to the system real great conflicts could be avoided. Nevertheless that also might give rise to some interesting stories. But nothing is so dramatic as the failures in other parts of the nuclear field. For instance the disaster of Chernobyl. That turned the awareness about the causes into practical international co-operation and confidence building measures in the fields of reactor operations, free communication about any safety event, related co-operation, mutual support, and prevention.

Today the States involved in nuclear power generation are giving up a part of their sovereignty e.g. to allow foreigners to collaborate on the safety culture of the operation of nuclear power plants by exchange programmes or by inspections. This is not only a political guided co-operation based on treaties, but also an exchange of personnel arranged e.g. by the World Association of Nuclear Operators.

The Acheson Lilienthal plan in theory, and the Euratom Treaty in practice solve the international divergence on essential points of the nuclear industry and the nuclear weapons. like safeguards, safety and security. Safeguards is not only intended to contain the horizontal proliferation from State to State, but also to contain the vertical proliferation of the weapons arsenals development. The two documents establish a supra-national authority for the responsible organisation. The European Union has with its "soft power", its "committology", and stimulating co-operative exchanges of people, knowledge, and last but not least also Esarda, etc created a real broad basis for progress. The Euratom Treaty stimulates an equal treatment to all civil nuclear activities, whether in nuclear weapon states or in non-nuclear weapon states. By knowing the amounts of nuclear material via the obliged declarations for military purposes, Euratom is also capable to limit the suspicions about the dimensions of that possible military use.

But the political and commercial concerns have limited the application or even prevented the fair play rules of the original concepts of the Euratom Treaty, because the direct advantages of a fair implementation of these rules have been underestimated.

In science and in daily life open communication builds trust, and stimulates more communication. It creates a sound atmosphere of confidence, not based on blind faith, but on real facts. Safeguards can help to avoid dangerously escalating conflicts. Euratom is an example how all States, with or without military use of nuclear material can be inspected today without discrimination[32]

## 9. Notes

[1] "Model Protocol Additional to the Agreement(s) between State(s) and the International Atomic Energy Agency for the Application of Safeguards", IAEA-document Infcirc/540(Corrected).
[2] High Level Expert Group Appointed by The European Commission Directorate General for Energy and Transport, "Review of the Euratom Safeguards Office, Main Report, Final Version", 15 February 2002.
[3] Jörn Harry, "Euratom Treaty, Safeguards and Non-Proliferation", ESARDA 2003,-th Annual Meeting, Symposium on safeguards and nuclear materials management, City Conference Centre Stockholm, Norra Latin Stockholm, 13-15 May 2003, Proceedings on CD-ROM.
[4] Smyth, Henry DeWolf; *Atomic Energy for Military Purposes, the official report on the development of the atomic bomb under the auspices of the United States Government, 1940-1945*; Princeton, Princeton University Press; 1945. In his foreword Major General L.R. Groves wrote: "All pertinent scientific information which can be released to the public at this time without violating the needs of national security is contained in this volume."
[5] Werner Heisenberg, "Über quantentheoretische Umdeutung kinematischer und mechanischer Beziehungen". Zeitschrift für Physik 33 (1925), S. 879-893.
[6] President Franklin D. Roosevelt presented in the State of the Union the Four Freedoms on 6 January 1941. In 1945 the Four Freedoms have been included in the Charter of the United Nations.
[7] Rwanda's Parliament has the highest percentage of women of any parliamentary body, the article appeared in the New York Times 26 February 2005.
[8] Kofi Annan Secretary-General, "In larger freedom: towards development, security and human rights for all", United Nations Document A/59/2005, 21 March 2005.
[9] "wealth of nations," The Ecomomist, 9 December 2006
[10] Antonio Damasio, "Looking for Spinoza", (Harcourt, 2003)
[11] This word has to be understood in this context as including all aspects of management. The sometimes more limited meaning of this word, like it is in French, had been the cause of a historical clash between General de Gaulle and Winston Churchill in the middle of 1944, around the invasion in Normandy and the control of the actions of British soldiers.
[12] Chester I. Barnard, Dr. J. R. Oppenheimer, Dr. Charles A. Thomas, Harry A. Winne, David E. Lilienthal, Chairman, "International Control of Atomic Energy, Prepared for the secretary of states committee on atomic energy by a board of consultants", Washington, D. C., 16 March 1946; http://learnworld/LW/LW.html
[13] David E. Lilienthal, Chairman of the Tennessee Valley Authority, who acted as Chairman of the consulting Board, Chester I. Barnard,President of the New Jersey Bell Telephone Company, Dr. J. Robert Oppenheimer, of the California Institute of Technology and the University of California, Dr. Charles Allen Thomas, Vice President and Technical Director, Monsanto Chemical Company, and Harry A.Winne, Vice-President in Charge of Engineering Policy, General Electric Company.
[14] On 3 December 1945, General Leslie Groves told the Combined Policy Committee and representatives of the Truman administration that the `real secret' of the U.S. success was obtaining a monopoly of high-grade uranium. Groves did not make this alleged 'secret' public until 1954; http://www.routledge-ny.com-ref-sforelations-october45.pdf.pdf.
[15] A private communication of mid 1996 with an experienced safeguards statistician resulted in the following scheme for allocation of safeguards resources:
The ideal way safeguards should go to allocate resources could be:
1 - what possibilities are there for acquiring unsafeguarded nuclear material.
2 - what can safeguards do to restrict, of monitor at least, this acquisition.
3 - what sources of safeguarded material are present and what possibilities
exist for undetected diversion.
In the end safeguards would shift to the "control" of feed materials such as natural uranium, with left over resources assigned to reprocessing, MOX and enrichment facilities. Countries with only reactors and LEU plants would hardly see a safeguards inspector.
[16] That moment has been quoted as being fixed as: only after the controls had been established. This is stated in an article by Randy Rydell, "Looking Back: Going for Baruch: The nuclear Plan That Refused to Go Away", Arms Control Today June 2006. It may have been introduced
[17] The report of 16 March 1946 states: The consideration of the steps of transition by which the special position of the United States may be relinquished involves quite other values. The sequence, the ordering, and the timing of these steps may be decisive for the acceptability of the internationa1 controls, but they will not affect its operability. Therefore, they present problems of negotiation between the nations within the UNO in the course of agreeing upon a charter for the Atomic Development Authority. Such problems of negotiation, in our opinion, are separable from the nature of the objective of the negotiation. They are problems which cannot be solved now, because they depend, among other things, on the motivation of the participating nations, on the political background of the negotiations, and on what may be conceived to be the separate, as opposed to the collective, interests of these nations.
[18] Winston Churchill said: Communist fifth columns are established and work in complete unity and absolute obedience to the directions they receive from the Communist center. ... the Communist parties or fifth columns constitute a growing challenge and peril to Christian civilisation. Source: Robert Rhodes

James, Winston S. Churchill: His Complete Speeches 1897-1963, (Chelsea House Publishers: New York and London), vol. VII, 1943-1949, pp.7285-7293.

[19] Winston Churchil lost the elections so during the meeting Clement Attlee became the British Prime Minister.

[20] http://www.dannen.com/decision/index.htm

[21] Winston Churchill, Triumph and Tragedy (Boston: Houghton-Mifflin, 1953) pp 669-70.

[22] Zhores A. Medvedev, "Stalin and the Atomic Bomb", This article was published in Russia in 1999, in Disarming the New World Disorder, on the occasion of the 120-th anniversary of Stalins birth on 21 December 1879.

[23] A.K. Kruglov, Shtab Atomproma, p. 18.

[24] At the Nuremberg Tribunal (an international military court of justice held in Nürenberg in Germany) sentenced 19 German war criminals, between 20 November 1945 and 1 October 1946.

[25] In particular the commission shall make specific proposals:... "(D) For effective safeguards by way of inspection and other means to protect complying states against the hazards of violations and evasions.

[26] A piece meal approach on separate items that can already be found in the Acheson Lilienthal plan, to consideration that never realised: Regional Fuel Cycle Centres (1975-1977), the International Fuel Cycle Evaluation (1977-1980), International Plutonium Storage, and International Storage of Spent Fuel (1978 1982), The IAEA's Committee on Assurance of Supply for states that fulfilled their NPT obligations 1980 - 1987 ending with the UN Conference for the Promotion of International Co-operation on the Peaceful Uses of Nuclear Energy, ElBaradei's proposal for multilateral co-operation on enrichment and reprocessing leading to the report on Multilateral Approaches to the Nuclear Fuel Cycle (2005) until the US proposal for an unverified Fissile Material Cut-off Treaty.

[27] Jörn Harry, "Euratom Treaty, Safeguards and Non-Proliferation", see note 3.

[28] Francis Fukuyama, "Trust, the social virtues and the creation of prosperity", Free Press Paperbacks division of Simon & Schuster INC. New York (1996), Chapter 22 the high-trust workplace pp 255 - 266.

[29] Mark Rhinard, “The Democratic Legitimacy of the European Union Committee System”, Governance: An International Journal of Policy, Administration, and Institutions, Blacwell Publishing, ISSN 0952-1895, Vol. 15, No. 2, April 2002 (pp. 185-210)

[30] Mark Leonard, "Ascent of Europe", Prospect March 2005, page 22 - 25; http://www.cer.org.uk/pdf/leonard_prospect_feb05.pdf.

[31] Bernard Bot, " Great game or European fair play? The European Union and the world in the 21st century", European Studies Centre, St Antony’s College, University of Oxford, Oxford, 1 December 2004.

[32] Jörn Harry, "FMCT Verification and Safeguards", ESARDA Bulletin Number 30, December 1999, p.1 - 11. Also available on internet from the ESARDA-website.

# New cooperation arrangements in States party to INFCIRC/193, a status report

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***Abstract:***

*Pursuant to the Safeguards Agreement published in INFCIRC/193, the IAEA and the European Atomic Energy Community (EURATOM) implement safeguards in connection with the Treaty on the Non-Proliferation of Nuclear Weapons (NPT) in the Non-Nuclear Weapon States (NNWSs) of the European Union (EU). The New Partnership Approach (NPA), agreed in 1992, resulted in closer and more efficient cooperation between the two organizations in their implementation of that Agreement. With the entry into force of the Additional Protocol to INFCIRC/193 (published in INFCIRC/193/Add.8), and the anticipated implementation of integrated safeguards in the EU NNWSs, new cooperation arrangements need to be established to ensure continued smooth and effective cooperation between EURATOM, the IAEA and the States concerned. With the implementation of the Additional Protocol, the IAEA is aiming to draw the broader conclusion that all declared material remains in peaceful use (based on the IAEA's conclusion of the non-diversion of declared nuclear material and of the absence of undeclared nuclear material or activities in the State) as a necessary pre-condition to the implementation of integrated safeguards in those States.*

*This paper reports on the current status of a number of initiatives and actions taken by the IAEA in cooperation with the European Commission's (EC) Directorate General for Energy and Transport aimed at a technically effective and cost-efficient implementation of integrated safeguards approaches in EU States party to INFCIRC/193. This paper describes the main practical roles expected of both organizations when this regimen is put into practice.*

**Keywords:** integrated safeguards; High Level Liaison Committee, principal matters

## 1. Introduction

Following the entry into force of the Protocol Additional (AP)[1] to the Agreement between the NNWSs of the EU, EURATOM and the IAEA on 25 October 2004, the IAEA devoted its efforts to drawing conclusions of the absence of undeclared nuclear materials and activities in each and every State party to INFCIRC/193/Add. 8. The Agency's ability to provide assurances on the absence of such undeclared activities created the potential for changes in implementation parameters and a reduction in verification effort for declared nuclear material, by combining, in an optimized fashion, all safeguards measures available to the Agency which achieve the maximum effectiveness and efficiency within the available resources in fulfilling the Agency's right and obligations as stipulated in Article 2 of the Safeguards Agreement in force. This combination of safeguards measures is known as "Integrated Safeguards" (IS). The Agency aims to reach a broader conclusion in relation to the non diversion of declared material and the absence of any undeclared material or activities for most EU States party to INFCIRC/193/Add.8 by the end of 2007, and to attain this goal for all States during 2008.

Once this essential prerequisite is reached, IS will be implemented following the mandate of the Board of Governors[2] which requested the Secretariat to proceed further with the implementation of IS following a conceptual framework developed by the Secretariat and endorsed by the Board. Against this background, the IAEA implements IS at the State level with an approach based on models for different types of facilities. In order to implement State-level approaches, it is necessary to agree with the European Commission (EC) and its Member States on the arrangements for a number of specific elements of the State-level, facility-type specific approaches. This paper describes the current status of the arrangements for the implementation of IS in EU States.

## 2. Principal Arrangements

After a period of relatively little contact at high levels, the Director General of the IAEA (IAEA DG) and the European Commissioner for Energy met in May 2005. The IAEA DG indicated that while the "New Partnership Approach" (NPA) agreed in 1992 served the Agency and the EC well, it was appropriate to revisit it in the light of new safeguards developments, such as the advent of IS. The EC Commissioner, in turn, agreed to renew high-level contacts between both organizations with a view to reaching agreement on the principal technical and administrative arrangements upon which the cooperation between the EC and the IAEA will rely. Following this, the then newly appointed Director General for Energy and Transport of the EC visited the IAEA in June 2006 and met with senior managers of the IAEA, discussed cooperation matters between the two organizations and agreed to start consultations to prepare a Liaison Committee [3] meeting at the high level (HLLC) in the near future. Shortly after this, a meeting was held in Vienna in the margins of the General Conference in September 2006. The Deputy Directors General of the EC and the IAEA Department of Safeguards co-chaired this meeting. To build upon the previous arrangements, both organizations agreed to focus on the implementation of integrated safeguards in the EU, taking into consideration the need to increase the effectiveness and efficiency of the safeguards system while assuring the independence of the IAEA's verification activities. At this meeting, it was recognised that there were a number of EU States not yet party to INFCIRC/193 where the broader conclusion has been drawn and IS approaches applied. It was agreed that these approaches should continue to constitute the basis of safeguards implementation in those States, even after accession to INFCIRC/193. Given these positive developments, it was agreed that technical meetings on principal matters should take place in October and November 2006 to discuss and draft the modalities for implementing integrated safeguards in the EU. Those modalities included, notably, the main elements of the State-level and facility type approaches developed by the Agency in line with the conceptual framework for integrated safeguards endorsed by the Board in 2002 and applied currently in many States throughout the world.

Following the undertakings reached at the higher level, representatives of the EC and the IAEA discussed a number of issues and exchanged information in technical level meetings. The IAEA introduced the approaches envisaged under integrated safeguards and provided the EC with copies of the Agency's model safeguards approaches for all major facility types. The specific elements of the State-level facility type specific approaches were discussed and agreed "ad referendum" in these meetings

The technical meeting concluded its deliberations and issued a report to the Liaison Committee with a proposal for coordinating activities when integrated safeguards are applied. The proposal included a table listing the activities anticipated to be carried out and included details regarding the inspection scheduling and verification activities, the communication channels and the containment, surveillance, non-destructive assay and remote monitoring measures. (See table "Safeguards Implementation in the European Union" in the Annex). The main concept of the arrangements included in the report to the Liaison Committee is that the EC and the IAEA will continue cooperating in traditional inspection scheduling and verification activities that will still be implemented under integrated safeguards. Examples of such activities include regularly scheduled inspections and DIV visits that are scheduled by the EC in cooperation with the IAEA and carried out jointly in accordance with the existing arrangements resulting from the New Partnership Approach. On the other hand, activities

resulting mainly from integrated safeguards, such as randomly scheduled inspections with short notice contemplated for some types of facilities, will be scheduled by the IAEA in cooperation with the EC. Randomly scheduled inspections without notification (also known as unannounced inspections) will be scheduled and carried out by the IAEA. In this case, the EC and the States concerned will be notified at the same time as the operator. Other field activities, such as complementary access (CA), will continue to be carried out as at present. The IAEA will continue to determine when CAs will take place, and will conduct the access within 2 or 24 hours as foreseen in the AP. Inspectors of the EC would participate as far as possible, as is the current practice.

The technical group also noted that the need for direct communications between the IAEA and the operators, is necessary to have a more dynamic and fluid communication regime in order to implement effective and efficient safeguards, although it was recognized that the safeguards agreement in force requires that all formal communications (such as 90(a) and 90(b) statements and inspection notifications) need to be channelled through the EC in Luxembourg.

The next step in the development of arrangements for the implementation of integrated safeguards was the resumption (inactive since 2001) of the Liaison Committee which met in the premises of the Delegation of the European Commission in Vienna and which took place on 18 January 2007. This meeting was a milestone in the relations between the EC and the Agency as it marked the resumption of regular communications between both organizations at this high level. This Liaison Committee included, for the first time, the participation of representatives of the States party to INFCIRC/193/Add.8, as stipulated in Annex III of the Additional Protocol. Representatives of Germany, Finland and Portugal participated, together with observers from their Permanent Missions to the UN. The HLLC reviewed the status of safeguards implementation and considered the report from the technical working group, among other topics. The HLLC endorsed the recommendations in the report, in particular the following:

- that IS will be introduced State-by-State on the basis of State-level safeguards approaches prepared by the IAEA, taking into account the Euratom entity. Implementation arrangements will be discussed in detail with the EC;

- that the principles of the New Partnership Approach (NPA) will continue to apply and will be the basis for future arrangements; keeping in mind that some adaptations to accommodate new developments would be necessary;

- that the modalities for randomly scheduled short notice inspections, including necessary supplementary measures, should be further discussed and agreed;

- that the limited number of cases where randomly scheduled inspections may need to be conducted without advance notification should be further discussed and agreed;

- that existing IAEA safeguards arrangements in the new EU Member States should be reviewed and appropriate coordination agreed (in the meantime, the IAEA's current IS approaches will continue to be implemented);

- that joint technical support activities, including containment/surveillance and NDA measures, and remote transmission of surveillance and other data, should be discussed and agreed, and common training programmes should be considered (the EC confirmed its commitment to provide the necessary technical equipment); and

- that Subsidiary Arrangements needed to be updated, but that such updating should await agreement on the practical aspects identified above.

The HLLC also agreed to the resumption of the deliberations of the Liaison Committee at the lower level (LLLC) and the 48th LLLC meeting took place in Luxembourg in February 2007. As indicated earlier in this paper, the IAEA DG considered that it was appropriate to revisit the NPA in a quest for efficiencies and enhanced cooperation. In line with this thinking, the IAEA drafted an adapted NPA paper for light-water reactors (LWRs) under IS and tabled it at the LLLC meeting as a first example of a model facility-type document to be used as a guideline for IS implementation. This paper included a number of elements both from the original NPA principles agreed on in 1992, and the conceptual framework endorsed by the Board of Governors in 2002. Additionally, the LWR-IS-NPA paper followed the IAEA model IS approach for LWRs. The IAEA intends that the adapted, principle NPA papers will form the basis to the cooperation arrangements for the implementation of IS in the EU.
Other agenda topics discussed in this meeting included the modalities for randomly scheduled inspections with short notice, the need to perform randomly scheduled inspections without advance notification, IS arrangements for new acceding EU Member States and joint technical activities. On this occasion, the EC delegation was not receptive to the new measures involved in IS. For instance, the EC has challenged the need for randomly scheduled inspections at LWRs, a central and core component of IS approaches implemented in other parts of the world.

With respect to the newer EU Member States, the IAEA emphasized that IS was already implemented in one such State before it joined the EU (Bulgaria) and was implemented in four others (Hungary, Latvia, Poland and Slovenia) before they acceded to INFCIRC/193/Add.8. In addition, IS will soon be implemented in some additional new EU States (e.g., IS will begin in Romania and Lithuania on 01 June 2007). The IAEA intends to continue implementing its IS approaches in the recently acceding States, even after their accession to INFCIRC/193/Add.8. For example, the IAEA is continuing to carry out random interim inspections in Poland and Slovenia on an unannounced basis, as had been previously agreed with the safeguards authorities in those countries.

It is clear that the EC would prefer to keep the number of IAEA unannounced inspections in the EU to a minimum since it is difficult for the EC inspectors to participate in such inspections. It has been agreed, however, that a limited number of unannounced inspections will be necessary to meet the IAEA's requirements under IS. The IAEA and EC are now in the process of identifying the specific facilities, in addition to those in States already having IS, where the IAEA will need to carry out unannounced inspections.

The IS approach for LWRs is particularly important due to the large number of LWRs operating in the EU. The basic IS approach is that the IAEA will carry out annual PIVs, as before, but will greatly reduce the number of interim inspections. Instead of quarterly interim inspections performed at each LWR on a rigid schedule, the IAEA will carry out a small number of randomly scheduled interim inspections. The randomly scheduled inspections could be either unannounced or on short notice[4]. The IAEA's approach is to carry out these inspections on an unannounced basis, in which case permanent surveillance would no longer be needed at spent fuel ponds in LWRs[5].A variety of case studies have shown unannounced inspections to be the most economical safeguards approach for LWRs. As noted above, however, the EC would prefer to minimize the number unannounced inspections in the EU and has proposed to provide a cost-free surveillance service to the IAEA in order that the IAEA random interim inspections in LWRs can be performed on a short notice basis, rather than unannounced. The practicality of this proposal will be tested at the Krsko LWR in Slovenia before any decision is taken.

With regards to surveillance, it was agreed that surveillance data will be shared equally among both organizations, subject to approved technical arrangements and the maintenance of the IAEA's independent conclusions.

Under IS the IAEA will implement a system of short notice random inspections (SNRI) for flow verification at all natural and low-enriched uranium conversion and fuel fabrication plants. A field trial is currently under way at the Juzbado fuel fabrication plant in Spain, and preliminary results are encouraging. Upon completion of the Juzbado trial, SNRI systems will be extended to other low-enriched uranium fuel fabrication plants in the EU, adjusted as

necessary to meet plant-specific conditions. SNRI systems are also being established in the natural uranium conversion and fuel fabrication plants in Romania.

Another field trial is testing the application of unannounced random interim inspections to the gas centrifuge enrichment plant in Germany. The IAEA has recently adopted a new safeguards approach for gas centrifuge enrichment plants that requires measures to detect undeclared feed material. It is hoped that the new type of inspection being tested at Gronau will prove to be an efficient means to achieve that capability.

## Final Remarks

The size and complexity of the nuclear activities in States of the EU demands substantial resources from the Agency, the EC, national authorities and operators to implement safeguards. The introduction of integrated safeguards in those States is an opportunity to increase the effectiveness of IAEA safeguards while reducing the burden on all parties involved. To make use of such an opportunity, it is essential that all parties cooperate under a new mindset to achieve the common goal of increased effectiveness and efficiency. The conceptual framework endorsed by the Agency's Board of Governors in 2002 and the guidelines and procedures developed in accordance with the framework requires a more dynamic and flexible approach to maintain the "broader conclusions" to be reached for EU States. This approach includes the optimum combination of all safeguards measures available to the Agency, including expanded provision of information, complementary access to sites and locations in the States, the use of advanced safeguards technologies and a greater use of short notice and unannounced inspections, among others. The current status of implementation of AP measures in the EU is satisfactory, but there is a need to enhance the use of advanced safeguards technologies and finalize arrangements to implement short notice and unannounced inspections so as to maintain and enhance the Agency's independent conclusions while reducing considerably the effort in the field deployed by the Agency, the EC, national authorities and last, but not least, the operators under integrated safeguards. The advantages to all parties involved are evident, but it is necessary to accept this new approach with flexibility, open-mindedness and full and fast responsiveness.

[1] Published in INFCIRC/193/Add.8 dated 12 January 2005

[2] Board of Governors meeting held on 18 March 2002 (ref GOV/OR/1044): The Conceptual
Framework for Integrated Safeguards. (ref. GOV/2002/8 )

[3] INFCIRC/193 Protocol Article 25

[4] For LWRs with fresh MOX, quarterly announced inspections or where adequate remote
monitoring systems are installed, quarterly evaluation of RM data will be performed

[5] Fuel transfer verification will be achieved with the use of regularly scheduled inspections

# Session 2

# Novel Technologies and applications

# Laser-based Techniques for IAEA Safeguards Detection and Verification Applications

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***Abstract:***

The safeguards system of the International Atomic Energy Agency (IAEA) relies on a wide variety of techniques and instruments to support its inspection efforts verifying States' compliance with their respective non-proliferation obligations. Laser technologies are becoming increasingly important, providing new and novel verification and detection tools for current and future safeguards activities. The IAEA has already had positive experience in its application of three-dimensional (3D) laser imaging to nuclear facility design information verification (DIV) to confirm the absence of undeclared structural changes between on-site verification activities. Another promising application under investigation is the verification of metal seals, widely used in many safeguards applications. By scanning a seal's unique microscopic surface structure, the inherent 'fingerprint' produced provides increased assurance against tampering and seal-counterfeiting.

In August 2006, the IAEA convened a Technical Meeting on the *Application of Laser Spectrometry Techniques in IAEA Safeguards,* where staff and experts evaluated verification and detection approaches utilizing laser-based techniques. The meeting participants generally agreed that laser spectrometry promises effective and cost-efficient alternatives to some existing inspection methods, as well as providing novel solutions for emerging verification and detection needs.

The following promising examples were cited in regard to safeguards applications:

- Laser-induced breakdown spectroscopy (LIBS) aims to complement environmental sampling techniques providing real-time and on-site elemental composition of solid materials and trace analysis.
- Optically stimulated luminescence (OSL) has the potential to confirm the operational history of the facility or building space in question and, the absence of undeclared activities.
- Tunable diode laser spectroscopy (TDLS) could be used to detect part-per-billion (ppb) concentrations of hydrogen fluoride (HF), which is a signature of uranium conversion and enrichment processes that require the use of UF6. This technique has also been proposed for on-site destructive analysis of UF6 samples and on-line enrichment monitoring.
- Light detection and ranging (LIDAR) systems are being considered for sampling the atmosphere above suspected locations to detect signatures of undeclared nuclear processes.

This paper will introduce the Novel Technologies Project, present the recommendations from the IAEA's technical meeting on laser spectrometry and present details of developments in laser-based techniques that are being evaluated or applied by the IAEA Department of Safeguards for detection and verification applications.

**Keywords:** safeguards equipment; laser techniques; DIV; enrichment measurements; undeclared activities.

## Introduction

The safeguards system of the International Atomic Energy Agency (IAEA) uses a wide variety of tools for the independent verification of nuclear materials and facility designs to verify States' compliance with their non proliferation obligations. Verification activities range from the measurement of declared inventories and flows of nuclear materials, design information verification (DIV), the search for undeclared nuclear materials to the confirmation of the absence of undeclared nuclear activities.

The IAEA is looking constantly to enhance its verification capabilities. Laser-based technologies are becoming increasingly important, providing new and novel verification and detection tools for current and future safeguards activities. Through technical developments, the number and variety of laser-based products has grown considerably. The application base has broadened with high reliability characteristics. Increasing numbers of laser components are commercially available at competitive prices. Potential recent applications encompass various safeguards implementation fields, facilitating new verification measures, complementing traditional instrumentation and/or replacing existing techniques by allowing shorter analysis time, improving measurement accuracy and allowing additional information depth.

## Laser for DIV Activities

Laser based technology is already used successfully for DIV which is an important activity in the overall approach towards the safeguarding of nuclear facilities. The main objectives of DIV are to confirm that existing facilities are used as declared by operators or States and to detect the presence of undeclared design features or hidden facilities, which may in turn indicate the presence of undeclared nuclear activities. The 3D laser range finder (3D-LRF) is already in routine use by the IAEA for DIV activities involving the largest reprocessing plant under safeguards [1]. The system is capable of confirming within an accuracy of millimetres that no structural changes have occurred since the previous scanning and, of highlighting changes that may have occurred, in particular to maintain continuity of knowledge (CoK) of the interiors of hot cells especially on various piping arrangements. For this purpose baseline scans — called reference scans — are performed during the plant construction; subsequent verification scans taken during periodic inspection activities are compared to the original references. The 3D-LRF system is comprised of a combination of commercially available 'off the shelf' hardware and specific software developed for IAEA application by the European Commission Joint Research Center. The complete system provides in a user friendly manner a very accurate and reliable tool, including features for authentication and encryption of data. The scanning data contain highly sensitive facility information and are maintained on-site under IAEA seals.

A further development of the 3D laser imaging system [2] combines high-resolution digital photography (visible, infra red), a geo-location system (e.g. GPS or Galileo) and an inertial navigation system with the laser system. This system will be used to perform outdoor DIV activities around complex nuclear sites. The system's software combines the data and integrates models using data acquired at ground level. The 3D model of the facility obtained could then be used by an inspector for DIV purposes on a laptop either on-site or the inspector could perform a virtual tour of the site at HQ. In addition, the 3D model could be used for inspection preparation and for training purposes.

## Laser for Unattended Item Identification

The IAEA has a vital need to uniquely identify and track movements of items such as various containers and fuel pins. The new safeguards approach for enrichment plants foresees the complete monitoring of feed and product material stored in $UF_6$ cylinders that enter and exit enrichment plants. Currently, these cylinders can only be tracked manually by checking the manufacturer's identification plate. These tags are easy to falsify and to replace. Therefore a credible method is needed to uniquely identify individual items. The monitoring of each cylinder is required to verify the absence of undeclared overproduction of material. The IAEA has successfully tested a laser based method that identifies individual $UF_6$ cylinders by the intrinsic spatial irregularities that are unique to each cylinder. This technique, developed by the European Joint Research Centre (JRC) at Ispra [3], would be coupled with video surveillance to provide a fully unattended system. The same technique would

maintain continuity of knowledge on cylinders which have been identified for verification until the quantitative assay is completed at a later stage. The IAEA has started to investigate the use of this technique in a fuel fabrication plant to monitor the inventory and flow of fuel pins in unattended mode.

## Laser for Surface Mapping

Containment verification techniques play an important role in IAEA safeguards approaches to maintain continuity of knowledge of the material contained in a specific container. Comprehensive containment verification needs to include not only the sealing arrangement, but also the integrity of the entire container surface as well as any welds to ensure that there was no penetration of the container, which could go undetected by the sealing system. A laser surface mapping instrument uses triangulation to build a precise 3-dimentional model of a container's surface [4]. This model can be used to identify slight imperfections that remain after a tampering attempt. The system consists of a laser scanner that creates a base line model of a container's surface during the initial container verification. This baseline model is compared with subsequent verification scans. Feasibility studies using containers which were modified and repainted in such a way that the naked eye could not see any difference, clearly revealed the "tamper attack" from the comparison of the surface maps. Using this technique to scan a weld surface can also provided a unique identification for a container.

## Laser Surface Authentication

Laser surface authentication (LSA) is a new technology, developed at the Nanotechnology Laboratories at Imperial College, London and the University of Durham, that may provide a new tool for item identification. LSA is based on a laser optical technique and uses the phenomena of laser speckle to recognize and extract the intrinsic 'fingerprint' within all material surfaces such as paper, plastic, metals and ceramics [4]. The LSA system will use a compact laser device with a focused laser beam to read this naturally occurring fingerprint as it is generated from the scattering laser light from the seal surface. This creates a unique identity code which can be stored in a database along with other information such as date, time of application, inspector identity, and location. The seal can be checked at any time by performing subsequent scans, which will automatically check against the existing stored 'fingerprint' and verify its authenticity. The materials of the metal surfaces alone are sufficient to supply a robust and potentially counterfeit resistant signature. Thus, the LSA system can provide a high level of security against seal counterfeiting compared to other technologies, at a fraction of the cost, since the seal itself, is already a low cost device, requiring no further modification.

At present all metal seals need to be detached for subsequent verification at IAEA, HQ, to ensure their authenticity. The LSA technique is however also applicable to in-situ authentication. It is anticipated that a compact field instrument will be developed for the inspector to verify metal seals in-situ. This has the added advantage of significantly decreasing the time required for seal verification.

The LSA technique has been proven in testing to be capable of identifying unique features of $UF_6$ cylinders, from a distance, in an enrichment plant. However, this solution had to be discarded because the technique was found only to be valid for proximity scanning.

## Laser Light Sources

The IAEA recently redesigned its electronic seal to further increase its security. One of the newly added features, meant to increase seal security, was the use of single mode fibre-optic cable. This cable, with a diameter of approximately 9 microns, makes tampering far more difficult. However, the smaller diameter increases loss of light due to slight misalignments inherent in cable connectors. A laser is used to overcome this loss and to maximize the light energy transferred through the cable. The open/closed status of the fibre optic is monitored by transmitting and receiving short light pulses at certain intervals. The additional light energy also makes it possible to increase the operational length of the fibre-optic sealing cable to more than 1000 m.

## Tunable Diode Laser Spectroscopy (TDLS)

TDLS is one of the most promising spectroscopy techniques for IAEA safeguards. The IAEA is studying various applications based on TDLS:

- On-site ambient air monitoring for HF traces detection.
- On-site destructive analysis of $UF_6$ samples.
- On-line enrichment and flow monitoring.

TDLS uses diode lasers that are tuned to access specific regions of the mid-infrared spectrum where most gases of interest have strong absorption while common gases, such as oxygen and nitrogen, do not have strong absorption. This offers extreme sensitivity to trace gases, being able to detect in the sub-ppb concentration range. The detection of HF could be indicative of conversion or enrichment related activities for all the processes that require the use of $UF_6$ and therefore is of prime interest for safeguards. Recently the detection of ppb concentrations of HF using a portable TDLS system was successfully demonstrated.

The tunable diode laser shines through a multipass cell, filled with the sample gas, and is collected by a detector on the other side. The detector then analyzes the absorption lines caused by the gas in the cell to identify the molecular composition of the sample. TDLS based detectors provide quick measurements with high spectral resolution in the infrared range.

TDLS instruments require no special safety measure for operation and are not intrusive. TDLS systems are portable and can be used both inside and outside a facility. In the latter case, the instrument can be mounted on automobiles, helicopters, aeroplanes or robots. It has been demonstrated that such systems could be applied to detect methane leakage from gas lines by utilizing an instrument mounted on a helicopter.

The precise measurement of the isotopically broadened absorption peaks of $^{235}U$ and $^{238}U$ allows for an estimation of the isotopic composition, i.e. enrichment, of a gaseous $UF_6$ sample.

Safeguards approaches in enrichment and conversion plants require bias defect testing of the isotopic concentration in $UF_6$. On-site NDA measurements using gamma spectrometry do not provide results with the required degree of accuracy. Taking samples for destructive analysis (DA) requires their handling and shipment. These activities are expensive and time consuming as they entail the off-site shipment of nuclear materials with the high associated costs. A feasibility study [6] has demonstrated the potential using the TDLS technique to determine the concentration of $^{235}U$ and $^{238}U$ in $UF_6$ on-site with an accuracy of better than 1% for $^{235}U$ enrichment. For this purpose a mid-IR laser with wide single mode tuning ranges, better than 4 $cm^{-1}$ of continuous tuning at 1290 $cm^{-1}$ and less tuning at 852 $cm^{-1}$ is required.

Using a TDLS based instrument, samples will be taken from the $UF_6$ cylinders and analyzed on-site with no need for sample shipment, allowing inspectors to draw safeguards conclusions in a timely manner.

In addition, the development of a TDLS instrument is foreseen to perform the on-line measurement of $UF_6$ parameters at enrichment plants. Such measurements would require bypass pipes which are transparent to laser light. For continuous operation, the effect of plating of $UF_6$ deposits on the inner side of the transparent pipe, which might change the optical behaviour, needs to be addressed.

## Optically Stimulated Luminescence (OSL)

Optically stimulated luminescence (OSL) uses the radiation induced charge trapping that occurs in many materials as a consequence of radiation dose accumulation. The accumulated charge can be retrieved by optical stimulation with a light source and the subsequent detection of the light emitted after the electron de-trapping. Therefore OSL can be used to detect the previous presence of radioactive materials and thereby contribute to the forensic analysis of past nuclear activities.

## Technical Meeting

In August 2006, the IAEA convened a technical meeting on the *Application of Laser Spectrometry Techniques in IAEA Safeguards* [7] within the framework of the project *Novel Techniques and Instruments for Detection of Undeclared Nuclear Facilities, Materials and Activities* [8]. The project
was established in 2005 to examine innovative technological solutions to strengthen the effectiveness and to improve the efficiency of safeguards. Experts from nine countries reviewed selected laser-based techniques for possible use in the following areas:

- On-site verification of safeguarded materials by on-line and off-line analysis and recording.
- On-site detection of undeclared activities and materials, including tools for complementary access (CA) and forensic inspections.
- Stand-off detection of undeclared activities and facilities.

The experts identified TDLS, cavity ring-down spectroscopy (CRDS), LIBS and LIDAR as the most promising laser technologies to address current and emerging safeguards needs. CRDS is an absorption spectroscopy similar to TDLS, where laser light travels over thousands of round trips in an optical resonant cavity having high reflectivity. Although CRDS might offer more sensitivity, its application in the field is limited by the specifics of the sample cell. Both techniques are non-intrusive and use relatively low power lasers.

Laser induced breakdown spectroscopy (LIBS) has the potential to perform elemental composition and trace analysis of solid materials to confirm past nuclear activities and the absence of undeclared activities, and is therefore considered as a possible screening device to reduce the number of environmental samples. Basically, LIBS is an atomic emission spectroscopy technique that utilizes a pulsed, well-focused laser to create a micro plasma on the sample surface. The resulting light emission spectrum of the decaying vapour plume has well known specific emission lines that are analysed by an integrated spectrometer.

Light detection and ranging (LIDAR) systems are being considered to search for undeclared nuclear activities from a distance by detecting volatile process signature compounds possibly released to the atmosphere. For safeguards purposes, differential absorption LIDAR (DIAL) is particularly interesting. This technology sends laser pulses tuned on two different wavelengths to the atmosphere — one specific to the strongly absorbing molecule, the other less absorptive as a reference — and then analyzes the intensity of light scattered back over time.

The principal recommendations arising from the technical meeting included the following:

- Pursue the development of an enrichment monitor based on TDLS for on-line and off-line measurements at enrichment plants.
- Pursue the development of CA instruments based on TDLS, CRDS and LIBS for the detection of gaseous and solid signatures and indicators of nuclear fuel processes, respectively.
- Study further the releases, probabilities and the behaviour of proliferation indicators and signatures in the environment in order to design detection strategies based on technologies already available, or in an advanced state of development.

## Conclusions

The IAEA's mandate to ensure that nuclear materials and activities are only used for peaceful purposes requires a wide variety of verification tools. The application of laser-based technologies in safeguards will play an increasingly important role in the provision of appropriate verification and detection tools for current and future safeguards activities. The technical meeting on the *Application of Laser Spectrometry Techniques in IAEA Safeguards* provided valuable guidance in this endeavour.

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ESARDA 29th Annual Meeting, Symposium on Safeguards and Nuclear Material Management, Aix-en-Provence (France), May 22-24, 2007

# Current CEA works on two laser spectrometry techniques as tools for detection of Undeclared Nuclear Activities

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# outline

Introduction : objectives of current CEA works

CRDS : cavity ringdown spectroscopy (and related techniques)

*Molecular absorption near-IR technique using tunable diode laser*

results : HF detection

perspectives

LIBS : laser induced breakdown spectroscopy

*multi elemental laser induced plasma spectrometry*

Results : classification of the aluminium alloys series

perspectives

Conclusion

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# Introduction

Objectives of current CEA works :

- Demonstrate the **trace level detectability of indicators** of undeclared nuclear activities.
- Develop & evaluate effectiveness of **on-site Chemical Traces Search Tools** for future safeguards inspections.

Fields of application currently investigated in a first step :

- Quick & sensitive detection of emanations associated with nuclear processes :
  - HF traces in atmosphere
- Discrimination of material compositions
  - Between different kinds of alloys : Aluminium
  - Between U concentrates coming from different origins

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# 1rst part :

# CRDS technique

*Cavity Ring Down Spectroscopy*

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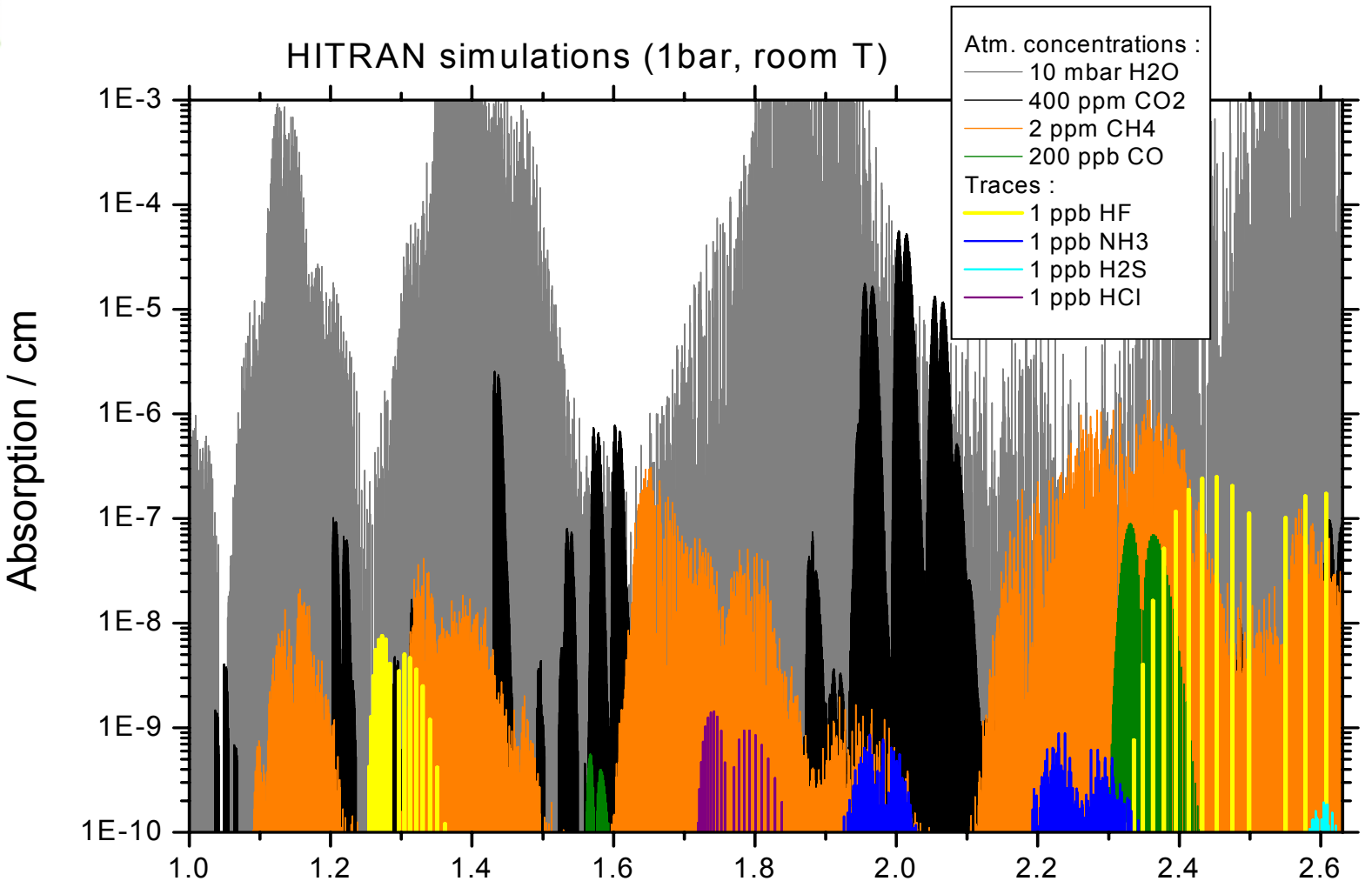
# Molecular transitions in the near IR

## 1 – 2.7 µm : room-T tunable diode lasers

*Small molecules*



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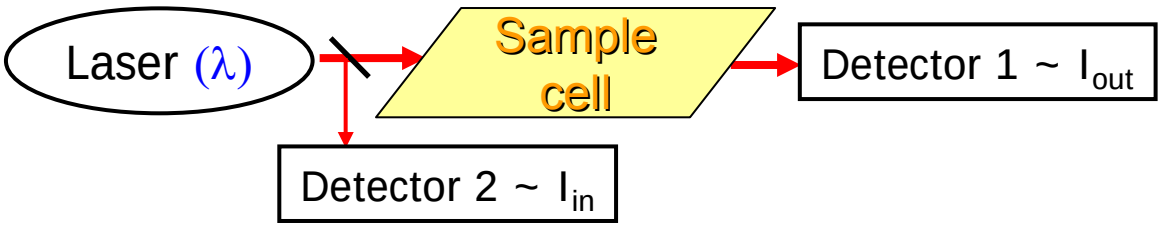
# Absorption spectroscopy




Beer-Lambert Law : $I_{out} = I_{in} \exp(-\alpha L)$

$\alpha L \ll 1$ ... $\Delta I / I = (I_{out} - I_{in}) / I_{in} \sim -\alpha L$

Higher sensitivity :

Increase L (path) .... and/or decrease $\Delta I$ (noise)

↳ High finesse optical cavity


R ~ 99,999%

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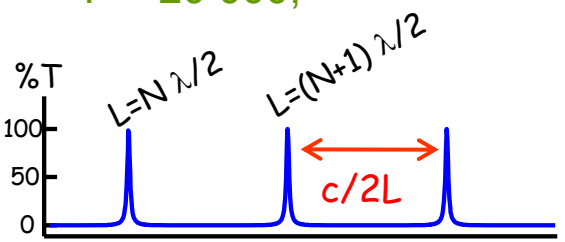
# Optical feedback

*Collaboration with D. Romanini, LSP/UJF, Grenoble, France*

High finesse cavity :

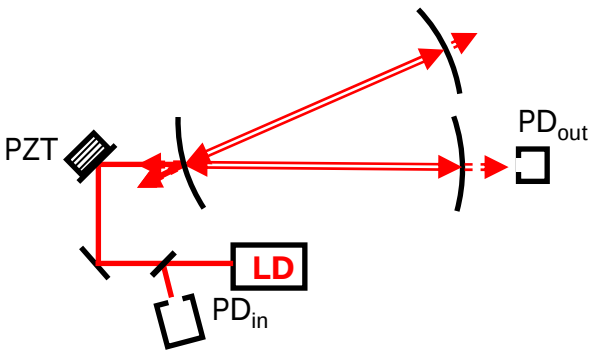
L = 1 m → ISL ~ 100 MHz,
R = 0.9999 $\Delta\nu$ ~ 10 kHz
$\tau_o$ = 33 µs, Leff = 10 km
F ~ 20 000,




DFB Telecom diode laser :

1 - 2 µm, InGaAs, 10 mW
spectral width : ~ 1 MHz
balayage 2 cm-1 ou 0.4 nm

**Optical feedback**
diode laser line narrowing
→ better cavity injection
Mode locking
→high cavity output




CRDS and CEAS with the same experimental set-up

CRDS : Cavity RingDown Spectroscopy
CEAS : Cavity Enhanced absorption spectroscopy

J. Morville & al., Appl. Phys. B 80, 1027-1038 (2005), J. Morville & al. Appl. Phys. B 78, 465-476 (2004)

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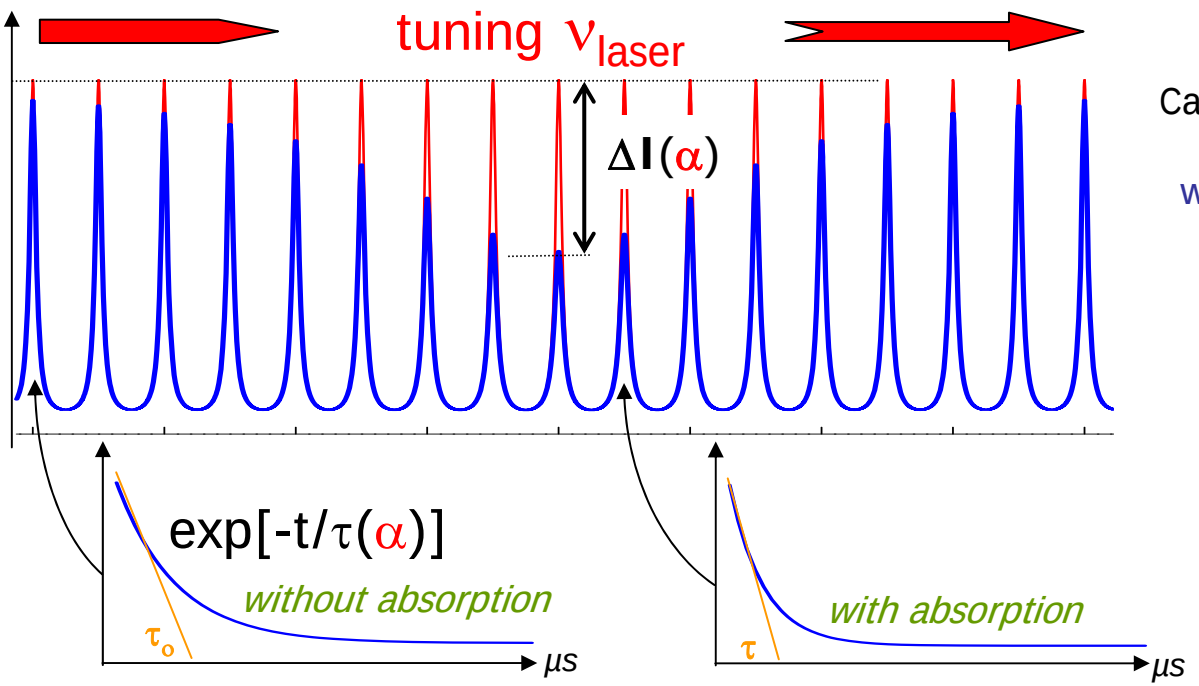
# CRDS and CEAS




Cavity transmission signal (CEAS) with and without Absorption

Ringdown signal (CRDS)

**CRDS :** $1/\tau = \overbrace{(A+T)\ c/L}^{1/\tau_o} + \alpha(\nu)\ c$

**CEAS :** $\Delta I/I = [\ 1 - R \cdot e^{-\alpha(\nu)\ L}\ ]^{-2}$

$\sim (2F/\pi)\ \alpha(\nu)\ L$

<u>absolute</u> measurement

A, R, T = mirrors Absorption, Transmission, and Reflectivity

L = cavity Length

Value of $\tau$ needed for Finesse

$F = \pi\ R\ /\ (1-R^2)$

# Advantages of laser diode absorption spectroscopy

## Optical spectroscopy

- Fast response : second range
- Sensitivity : (sub)ppb range ($\alpha$ = $10^{-9}$ to $5.10^{-11}$ $cm^{-1}$ $Hz^{-1/2}$)
- Linearity over a large dynamic range
- Selectivity (isolated absorption lines needed...)
- Non destructive measurements... (in-situ or remote)
- using diode lasers : compact, robust, easily tunable, low consumption, reasonably priced ...

⇒ *promising for field applications !!*

Absorption spectroscopy : direct measurement of the $\alpha$ (absorbance) of an absorption line, which is linearly proportional to the molecular concentration... : absolute measure

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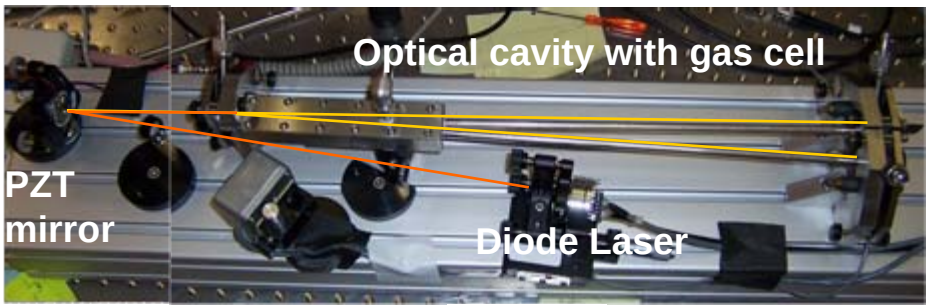
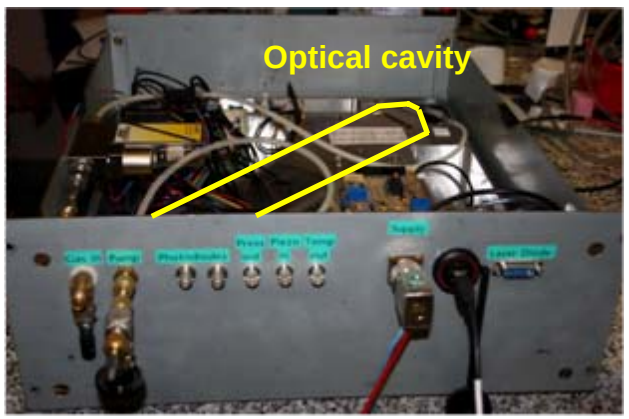
# CRDS and CEAS experimental set-up

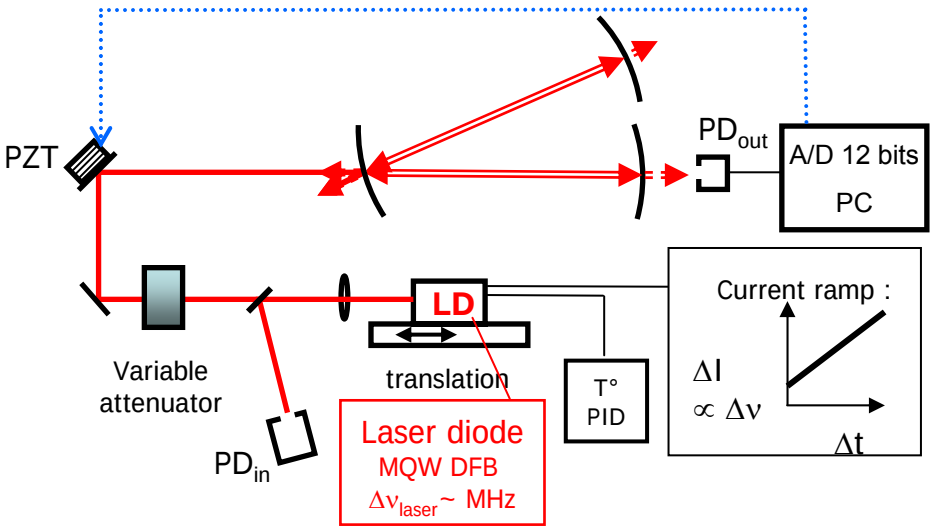
## *Belongs to TDLS techniques*




**To Photodiode**
**CRDS : ringdown**
**CEAS : I(ν)**




Rackable OF-CEAS prototype
*LSP/UJF, France*



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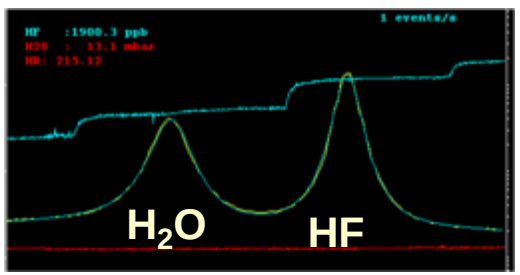
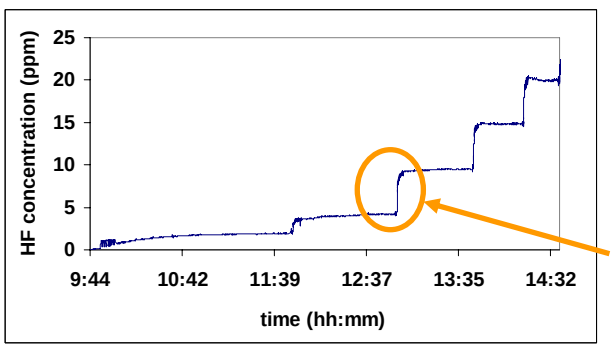
# Application : HF calibration

## *Collaboration UMR CEA-Université de Franche Comté*

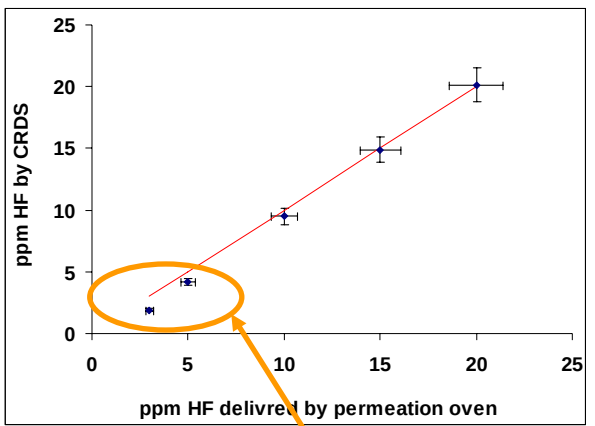
HF gas permeation oven (Calibrage)
Calibrated HF flux (±10%)
Range 1-20 ppm and 45-800 ppb
Primary HF flux 50 ml/min in $N_2$
Further dilution with $N_2$

CRDS : check the spectroscopic parameters of the HF 1312 nm molecular transition




Response time Of the CRDS system : 2 sec




Max gas flow : 0.2 l/min → adsorption

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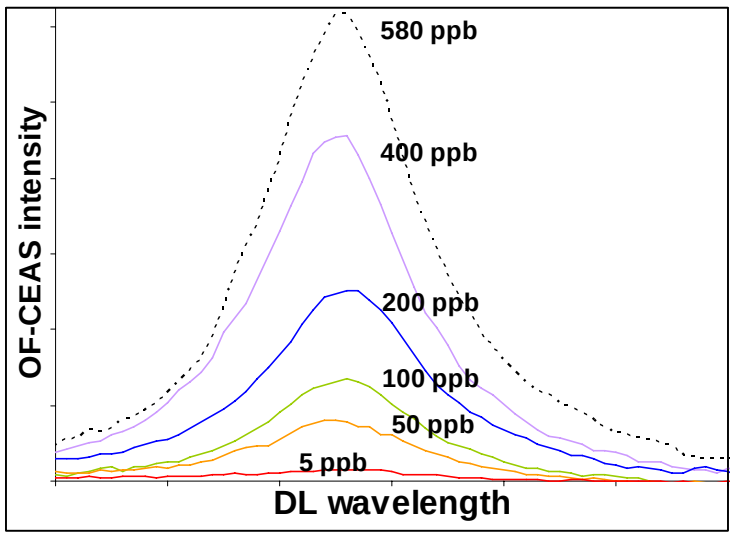
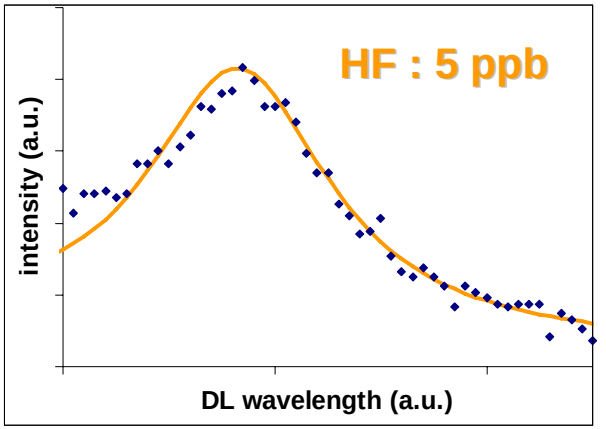
# Application : HF detection limit

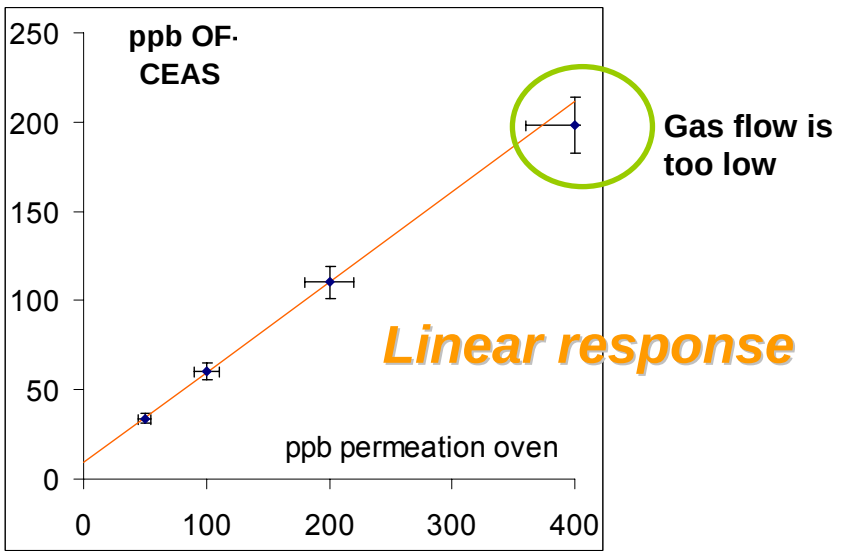
## OF-CEAS spectra : *1.6 Hz*




## Sensitivity

**Residual HF in the optical cavity**



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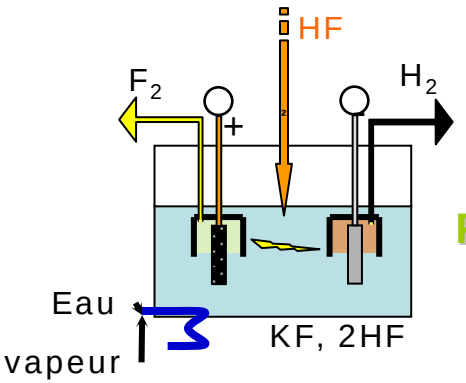
# CRDS/CEAS : field measurements

close to $UF_4$ fluoration plant, Pierrelatte, France
→ $UF_6$ production, before enrichment process

2007




$F_2$ production

Reaction chamber
$F_2$ + $UF_4$


Including instrument comparison in 2 steps :

1. SAW (surface acoustic wave) + OF-CEAS + chemical titration
2. TDLS (Canberra) + OF-CEAS + chemical tritation

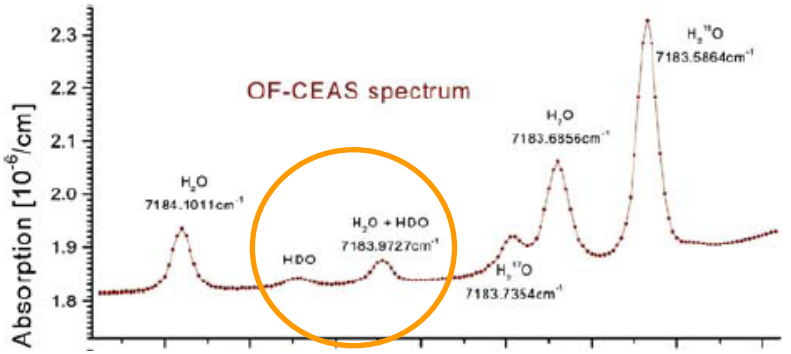
2008

HDO trace measurement

1. In laboratory
2. nearby a heavy water production facility

# CRDS/CEAS : field measurement

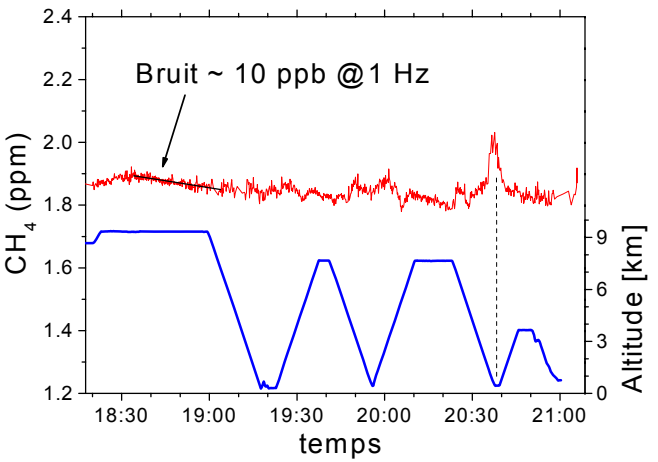
**Water isotopes and methane are measured in the stratosphere by OF-CEAS by UJF/LSP (France) and U. of Groningen (NL)**




*ERT Kerstel & al., Appl. Phys. B85, 398-406, 2006*


*August 2006, Burkina faso*

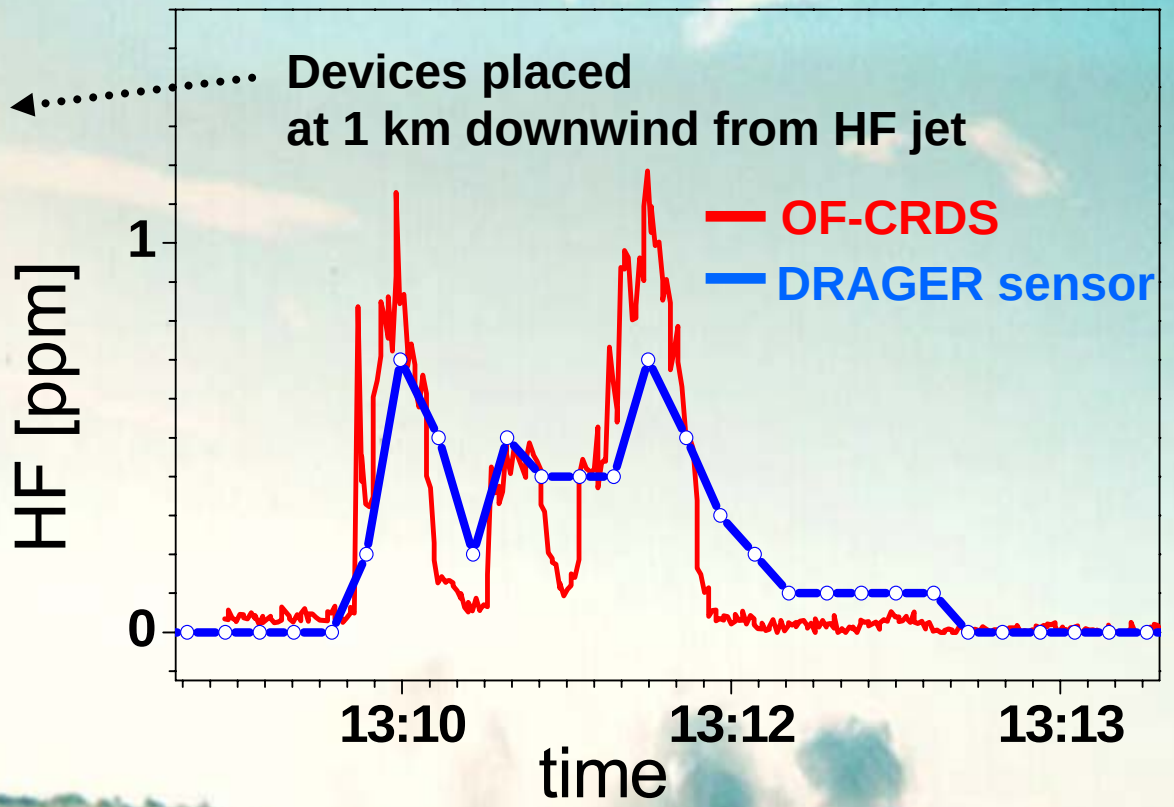
*Appl. Phys. B 83 (2006) 659*




*2005 troposphere flight with DC8 NASA aircraft*

# OF-CRDS : field measurements

*URAHFREP/CEA-Grenoble ( 2000, Porton Down - UK )*




**Authors :**
**D. Romanini & coworkers**
**UJF / LSP France**

**HF +$NH_3$**

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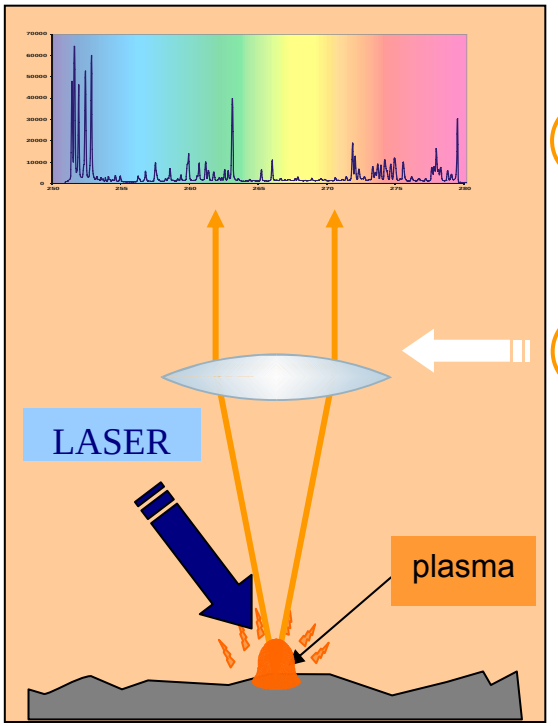
# 2nd part :

# LIBS technique

*Laser Induced Breakdown Spectroscopy*

# LIBS basic




5 Dispersed light spectra is detected (a few 100 ns to µs scale)

4 Light emission from excited atoms and ions is collected

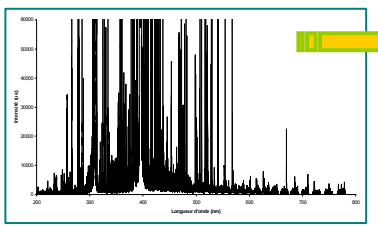
3 Plasma creation, expansion and cooling down

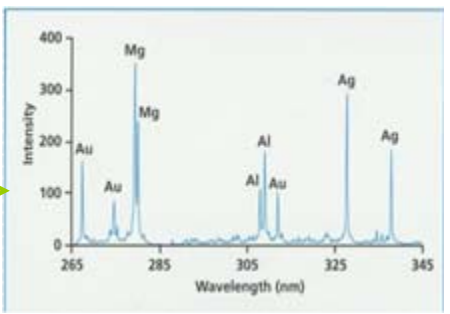
2 ng to µg sample vaporization

1 5 ns laser beam focused onto sample surface : ~10 GW/cm²

**all elements are detectable !**

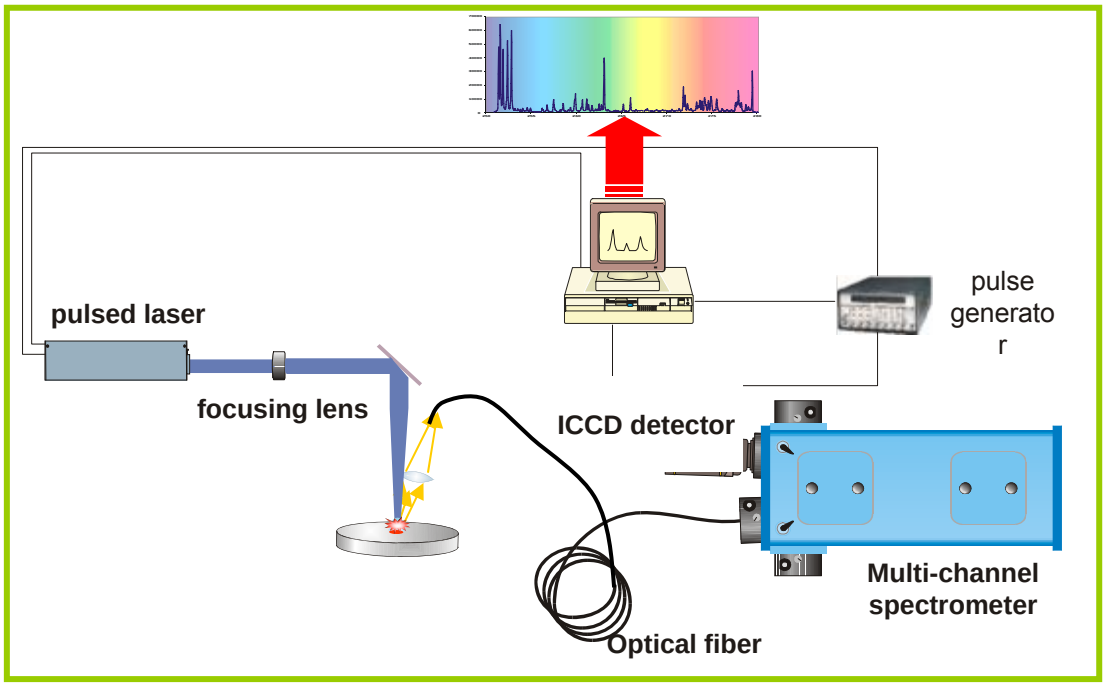
**Spectra analysis : multi-elemental composition and concentration**






*UV-visible spectra*

# LIBS experimental set-up




transportable system

collaboration between CEA and IVEA
(french start-up company)

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# LIBS : advantages

- **Main characteristics of the LIBS technique**
  - **Any material, solid, liquid, gas, aerosol**
  - **Simultaneous multielemental detection and analysis**
  - **Real time acquisition**
  - **No pre-treatment of sample**
  - **Suitable for in situ measurements, without sample cell**
  - **Micro-destructive technique *(ng to pg)***

- **A recent breakthrough in component development**
  - **Portable high resolution spectrometers, with a broad spectral range allowing the detection of all the elements** (from 200 to 900 nm)
  - **Compact solid state lasers**

**LIBS is well adapted for field applications**

» **in environmental monitoring and control process**

» **in security and space based research**

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# LIBS application : aluminium alloys identification

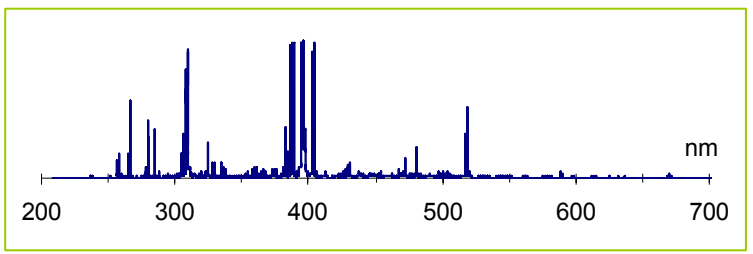
## Aluminium 7000 serie : highly present in centrifuge plant

(wrought) aluminium alloys classification :

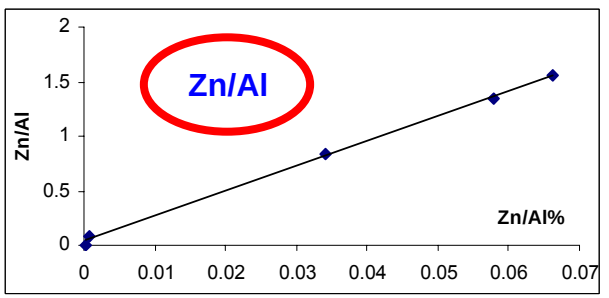
| serie | 1 | 2 | 3 | 4 | 5 | 6 | 7 |
|---|---|---|---|---|---|---|---|
| alloy | pure Al | Al **Cu** | Al **Mn** | Al **Si** | Al **Mg** | Al **Mg Si** | Al **Zn** |
| | Al > 99 % | Al ≥ 95 % | | | | | |

Aluminium alloys are fully defined with 8 elements :
Cr, Cu, Fe, Mg, Mn, Si, Ti, Zn

Classical classification method : elemental composition




Transition intensity $\alpha$ element concentration



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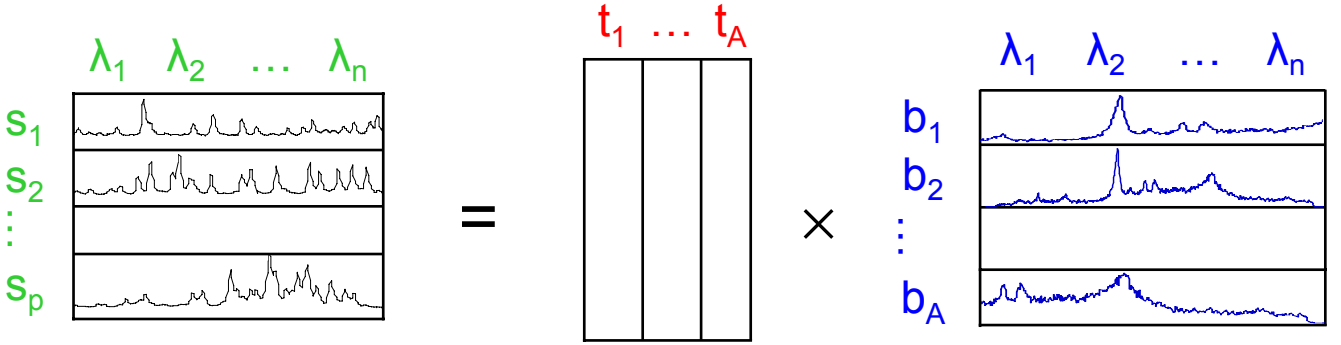
# Al alloys identification : chemometrics methods

Chemometrics : statistical spectra processing
extracting relevant informations from the experimental datas (spectra)

## PCA : Principal Component Analysis

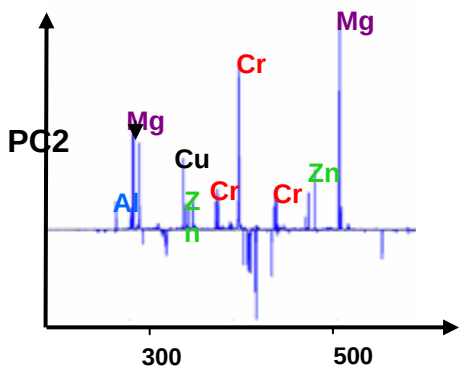
*Spectra decomposition according a new base*




***Spectra*** *in the base of wavelengths*

*Coordinates of spectra in the new base (**scores**)*

*Vectors of the new base (**loadings**)*




*Loading example*

Loading : spectral regions which have the largest variance

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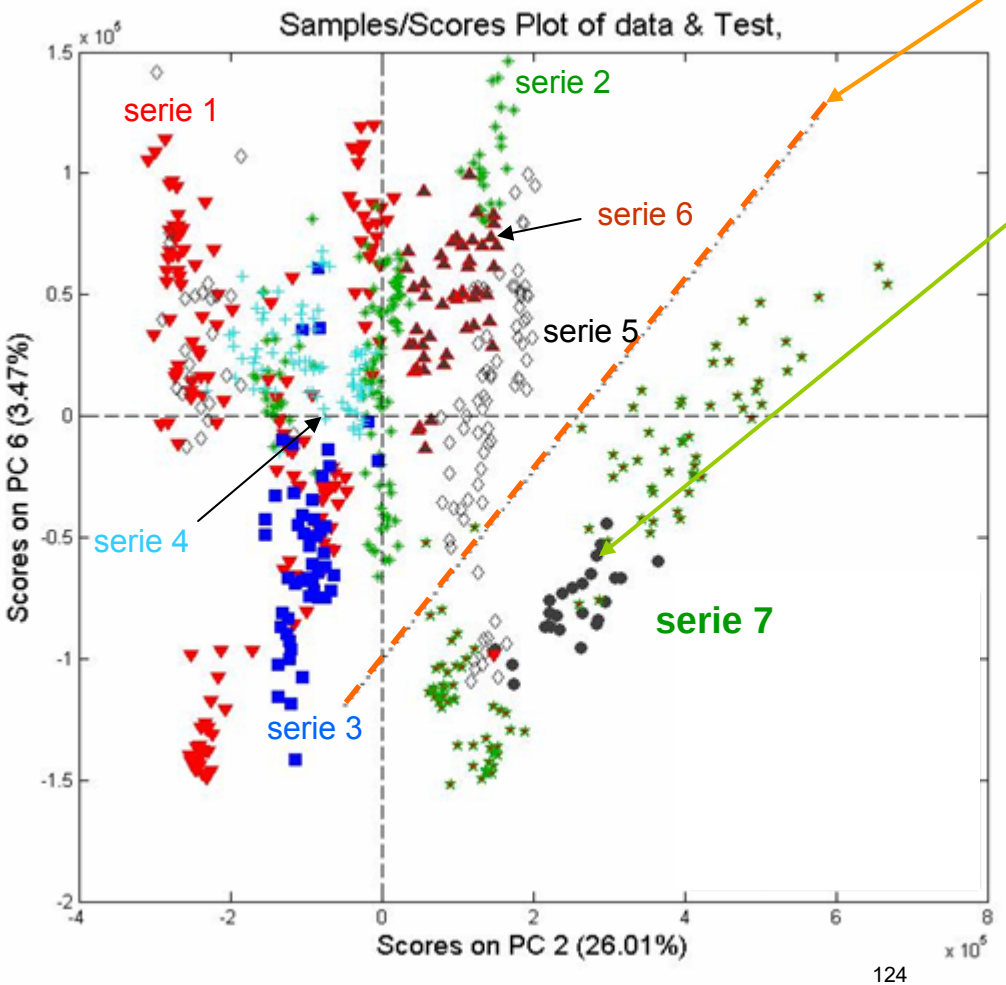
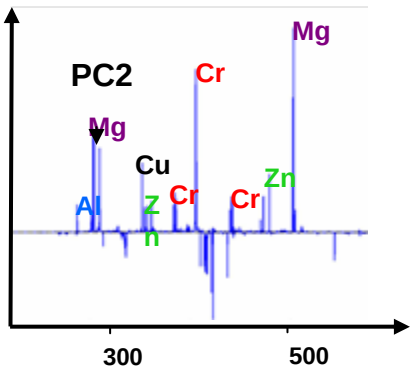
# PCA result : 7th Al serie identification




Line separates the 7th Al serie

Test sample (7th Al serie)




PCA results : graphs

SIMCA method

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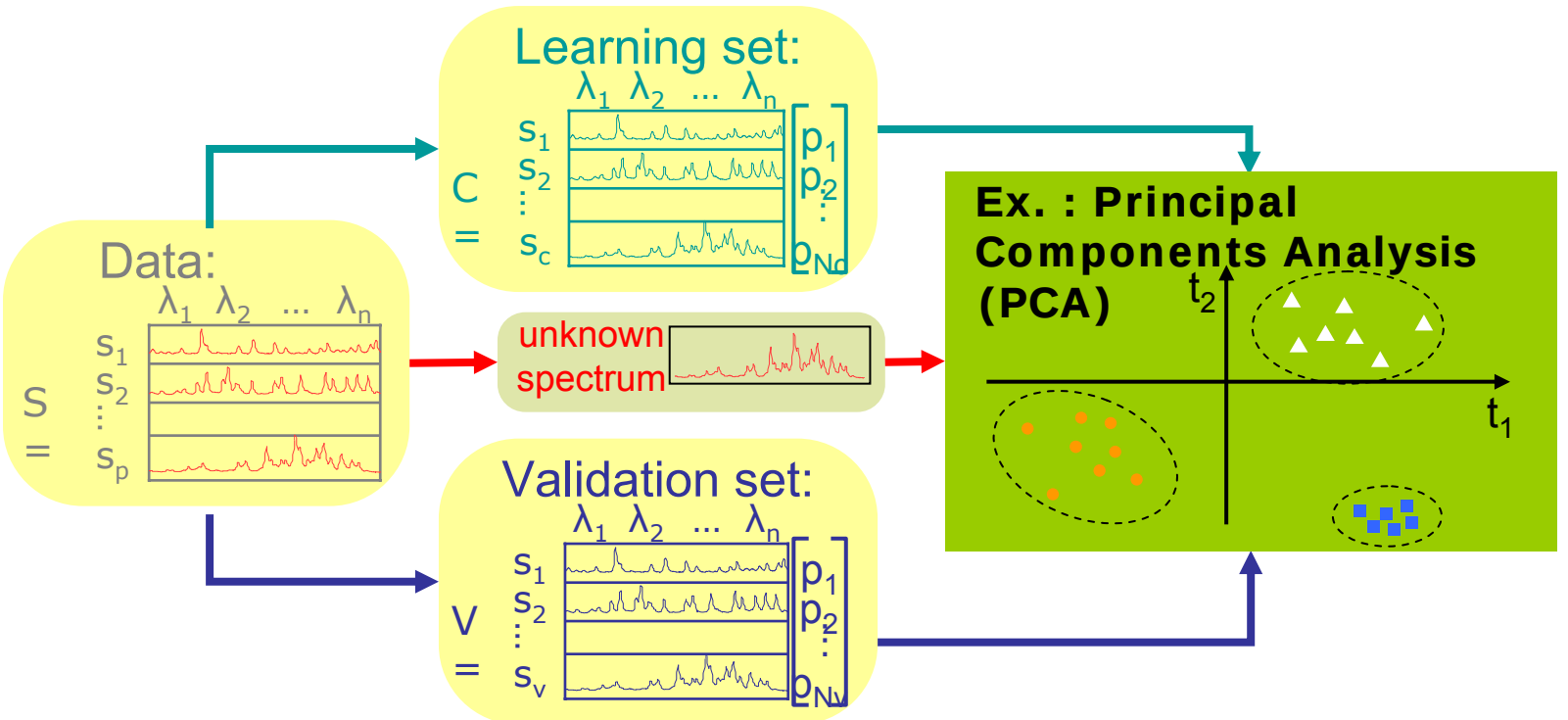
# SIMCA

## SIMCA : Soft Independent Modeling of Class Analogy

**Learning** : a PCA model is calculated for each class with a set of known spectra

**Prediction** : the unknown spectra is allocated to the "nearest" class



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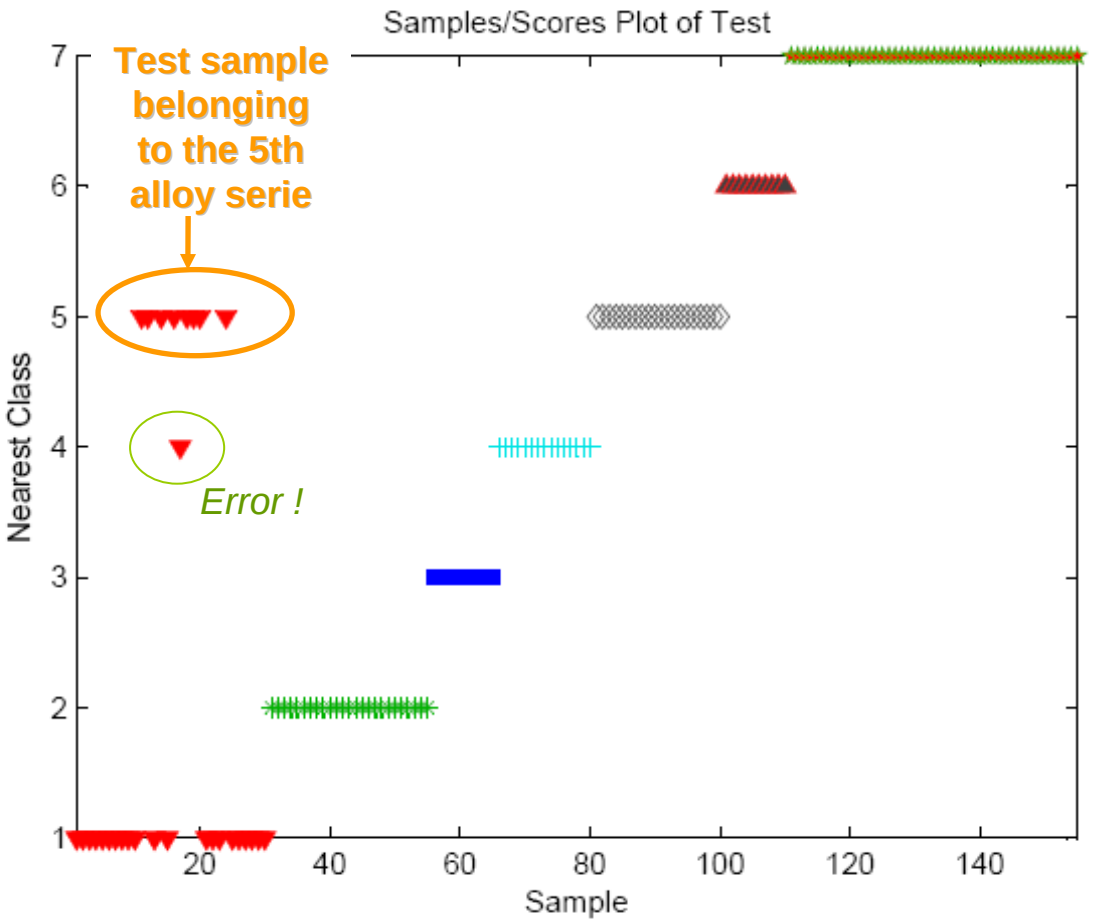
# SIMCA results : Al alloys series identification




*After learning, 98% of the spectra are successfully classified despite the large variance inside each Al class*

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# Next …. chemometrics

LIBS + chemometrics : *efficient method for classification*

Discrimination of the yellow cake (uranium concentrates) origin

Yellow cake composition impurities depends on the mine, the extraction technique ….

| impurities | Units | origin 1 | origin 2 | origin 3 |
|---|---|---|---|---|
| Mo | µg/gU | 1 | 750 | <0,5 |
| W | µg/gU | 2 | 153 | <0,5 |
| Si | µg/gU | <20 | 70 | 130 |
| Al | µg/gU | 570 | 50 | 260 |
| Cr | µg/gU | 21 | 10 | 8 |
| Cu | µg/gU | 3 | 4 | 1200 |
| Fe | µg/gU | 7400 | 100 | 760 |
| Mg | µg/gU | 110 | 21 | 250 |
| Ni | µg/gU | 8 | 3 | 2 |
| Ca | µg/gU | 1300 | 40 | 3300 |
| K | µg/gU | 280 | 6 | 14200 |
| Na | µg/gU | 2500 | 42 | 5700 |
| Pb | µg/gU | 6 | <6 | 12 |
| Th | µg/gU | 1350 | <0,5 | <0,5 |
| Zn | µg/gU | 1 | 2 | 50 |
| Zr | µg/gU | 6 | 40 | <0,5 |

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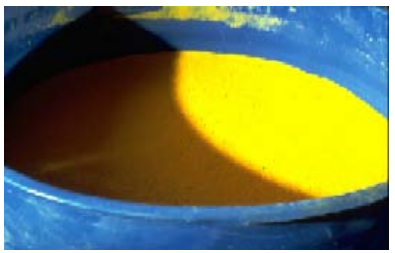
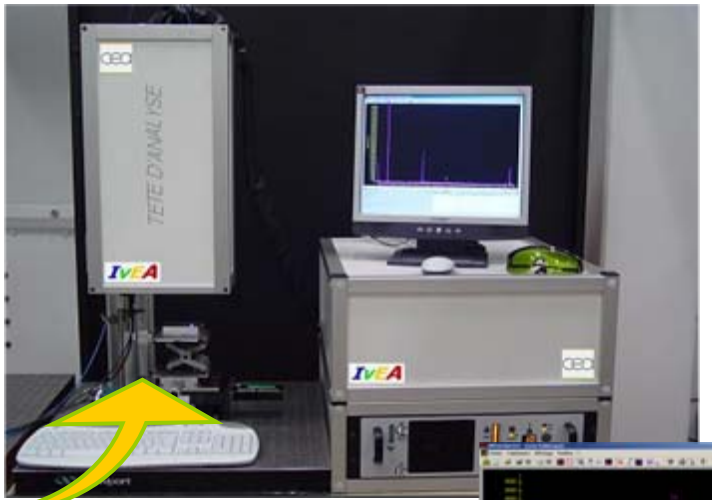
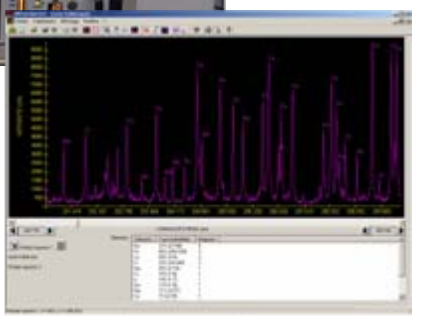
# Next …. LIBS



« yellow cake »

Air tight chamber



2007

U traces LIBS detection on a handling vehicle air filter

*Vehicle air filter is natural particle concentrator*

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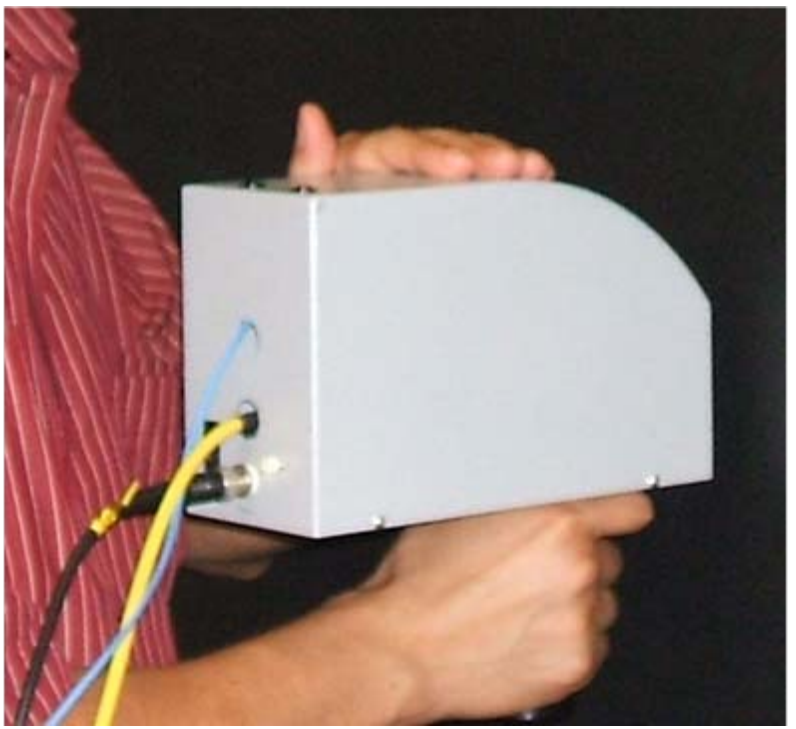
# Next …. portable instrument

Pistolibs : laser light and signal are transported through optical fibers

2008



***IVEA prototype***

# Conclusion

## .... *CRDS* ....

**Short term 2007 : CRDS/CEAS technique adapted to IAEA safeguards ?**

- **HF sensitivity : sub ppb ?**
- **field measurement + interference with industrial molecules ?**
- **comparison with TDLS and other techniques**

**Mean term 2008**

- **ready for a commercial instrument ?**
- **HDO measurement**

## .... *LIBS* ....

**Short term 2007**

- **Discrimination of the uranium concentrates origin** (LIBS+chemometrics)
- **LIBS detection of uranium traces on an engine air filter**

**Mean term 2008**

- **Complementary work to build a portable field prototype**

# STABLE NOBLE GAS ISOTOPES FOR STRENGTHENING NUCLEAR SAFEGUARDS
## The measurement point of view

**Y. Aregbe*, S. Valkiers*, K. Mayer**, M. Varlam*, R. Wellum***

*European Commission – Joint Research Centre
Institute for Reference Materials and Measurements (IRMM)
Geel, Belgium
** European Commission – Joint Research Centre
Institute for Transuranium Elements (ITU)
Karlsruhe, Germany

***Abstract:***

*The IAEA is currently investigating new technologies to strengthen the effectiveness of its inspection and verification activities. In particular, the IAEA is looking for new methods and instruments, which are applicable to the detection of undeclared nuclear activities and facilities. Recently the IAEA Department of Safeguards held a Co-ordinated Expert Meeting on Noble Gas Monitoring at IAEA Headquarters in Vienna to review the applicability of noble gas sampling, analysis and monitoring for IAEA safeguards. A feasibility study was performed at the EC-JRC - Institute for Reference Materials and Measurements (IRMM) in cooperation with the EC-JRC-Institute for Transuranium Elements (ITU) investigating the potential of identifying reprocessing activities by means of isotopic measurements of xenon and krypton. The KORIGEN code and the SCALE program were used to calculate the nuclear inventory of spent fuel for two scenarios involving high burn-up fuel as well as for two scenarios that could be used for production of weapons-grade plutonium (low burn-up).*
*Noble-gas isotopes are produced during the fission process: their isotopic composition is largely different from those of natural noble gases. Consequently, significant isotopic alterations are caused by dilution of the released noble gases with those in the atmosphere. This dilution process was "simulated" for noble gas generation due to high and low burn-up reactor operating scenarios by examining the isotopic alteration of xenon at different blending ratios.*
*The potential and limitations from the analytical measurement point of view, considering routine and reference measurements, to measure these changes in stable xenon and krypton isotopic ratios were discussed. A potential laboratory network approach yielding reasonable cost-benefit quality assurance was suggested. The conclusion was reached that characteristic xenon and krypton signatures originating from irradiated fuel are a promising additional tool for confirmation of operation declaration of fuel reprocessed at large nuclear reprocessing plants. Within some constraints the accumulated information gained from xenon signatures could also provide valuable information for nuclear safeguards verification of undeclared reprocessing activities, supplementary to radiometric measurements of $^{85}Kr$.*

**Keywords:** Noble gas signatures; isotope ratio measurements; nuclear safeguards; quality control

## 1. Introduction

Recently, concerns in view of changes in the nuclear programme of some countries that have signed the additional protocol and/or the INFCIRC 66 type safeguards agreement became of huge public interest. The media and the press reported on IAEA's negotiations with political leaders and on scientific expert opinions. Once more the question of the purpose of nuclear weapons in the nuclear weapon states and the commitment of the non nuclear weapon states to never assemble such weapons of mass destruction were in the centre of public attention. The IAEA in its role of

safeguarding nuclear activities and nuclear material so that it is only used for peaceful purposes faces technical and political challenges to ensure that there is and that there will be no deterrence of weapons grade nuclear material in the future. To this end the lAEA is currently investigating new technologies to strengthen the effectiveness of its inspection and verification activities. In particular, the IAEA is looking for new methods and instruments, which are applicable to the detection of undeclared nuclear activities and facilities. Noble gas monitoring has already been proposed in the past for the detection of undeclared reprocessing activities. The characteristic signatures of reprocessing of the released fission noble gases were due to the level of technology considered not applicable for IAEA needs at that time. Improvement in analytical techniques over the last 10 years encouraged the IAEA to look again into the matter of noble gas monitoring as safeguards tool. Therefore in 2005 the IAEA organised a technical meeting to review the applicability of noble gas sampling, analysis and monitoring for IAEA safeguards. The meeting participants discussed the applicability of noble gas monitoring, sampling and analysis for strengthening the IAEA safeguards in view of two basic objectives for short-range detection of noble gases for detecting undeclared activities within the vicinity of declared facilities and long range detection of undeclared activities. For short-range detection the isotopic signature of released fission off noble gases, particularly of stable isotopes, was identified to be one major point of interest. The European Commission Joint Research Centre Institute of Reference Materials and Measurements (EC-JRC-IRMM) was invited to this technical meeting due to its well-known expertise in the field of gas isotope ratio measurements. A feasibility study on the potential application of stable isotope ratio monitoring of noble gases was carried out in cooperation with the EC-JRC-Institute for Transuranium Elements (EC-JRC-ITU) to provide recommendations to the IAEA for the potential application of stable noble gas monitoring

## 2. Production of stable noble gases

Stable xenon and krypton isotopes would be primarily produced during fission in the core of a reactor. They are either generated as direct fission fragments or as daughter nuclei of beta-disintegration. Therefore they show a greater abundance of heavier isotopes compared to atmospheric xenon and krypton coming from the neutron excess of the initial fissionable nucleus. In addition the cumulative fission yields of the light isotopes are suppressed by the shielding in fission product beta-decay chains by very long lived precursors such as $^{129}$I (107 yr half life). Noble gases are released during dissolution of the fuel batch and have a characteristic isotope signature depending on the history of the fuel being reprocessed.

## 3. Objective for stable noble gas application

At the technical meeting the IAEA express the need for short range detection to detect undeclared production and reprocessing of one SQ (8 kg) of weapons grade $^{239}$Pu in one year by a typical burn-up of 1000 MWd/MTU. To make a feasibility study on the potential of detecting the production of 1SQ Pu by means of stable noble gas measurements probable proliferation sub-cases on reprocessing of the fuel were identified.

### 3.1. Proliferation sub-cases

For the first reprocessing sub-case the dissolution of a large batch was considered over a year. The second sub-case assumed a proliferation scenario of weekly dissolutions. The case of daily dissolutions of small amounts of the fuel batch was considered as one of the most realistic scenarios for clandestine Pu production. Daily dissolution would admittedly bear the risk for a proliferator to release characteristic signatures for environmental sampling on a daily basis, but those signatures would be less significant with smaller alterations in the isotopic composition and thus more challenging to be detected by IAEA safeguards. Furthermore for all the 3 sub-cases a dilution factor in the reprocessing plant's stack of $10^5$ m$^3$/h was thought to be representative.

## 4. Feasibility study

A feasibility study was carried out to investigate whether the isotopic alterations in Xe and Kr collected in stack would be from a mass spectrometry point of view still measurable one hour after release. The KORIGEN code and the SCALE programme were used to calculate the fissiogenic noble gas inventories for 4 different reactor operating scenarios. Two for electricity generation of 2 different reactor types and 2 for clandestine Pu production in the same reactors:

- Scenario A: PWR, UOX 3.5 % $^{235}$U enrichment, high burn-up
- Scenario B: PWR, UOX 3.5 % $^{235}$U enrichment, low burn-up
- Scenario C: CANDU, natural U, high burn-up
- Scenario D: CANDU, natural U, low burn-up

There is a correlation of released noble gases to the magnitude of Pu production of a specific reactor operating scenario. The low-burn up scenarios account for the production of weapons grade Pu of 98% enriched $^{239}$Pu [1].

## 4.1. Signal to noise ratios

Once released stable Xe and Kr mix with atmospheric Xe and Kr. More xenon than krypton is present in the fission off gases. Since xenon is less abundant in air than krypton the feasibility study focused on the xenon fissiogenic inventory. Table 1 shows the parameters of the different scenarios and the amount of fission Xe produced due to the 3 proliferation sub-cases.

| | Scenario A | | | Scenario B | | |
|---|---|---|---|---|---|---|
| **Reactor type:** | | | UOX-PWR | | | |
| **Burn-up** | | 33 GWd/MTU | | | 1 GWd/MTU | |
| **Fuel** | | | UOX-3.5 % $^{235}$U enrichment | | | |
| **Irradiation time:** | | 3 years | | | 1 month | |
| **Cooling time:** | | 3 years | | | 1 month | |
| **air flow rate in stack:** | | | $10^5$m$^3$/h | | | |
| **proliferation subcases:** | 10x / year ~per 100kgU | 1x / week ~per 20kgU | 1x / day ~per 3kgU | 10x / year ~per 100kgU | 1x / week ~per 20kgU | 1x / day ~per 3kgU |
| **Fission Xe in g:** | 541 | 108 | 16 | 21 | 4 | 1 |
| **Fisson Xe : atm. Xe after 1 h:** | 1 : 0.1 | 1 : 0.4 | 1 : 3 | 1 : 2 | 1 : 10 | 1 : 69 |

| | Scenario C | | | Scenario D | | |
|---|---|---|---|---|---|---|
| **Reactor type:** | | | CANDU | | | |
| **Burn-up** | | 9 GWd/MTU | | | 750 MWd/MTU | |
| **Fuel** | | | natural uranium | | | |
| **Irradiation time:** | | 1 month | | | 1 month | |
| **Cooling time:** | | 1 month | | | 50 days | |
| **air flow rate in stack:** | | | $10^5$m$^3$/h | | | |
| **proliferation subcases:** | 10x / year ~per 100kgU | 1x / week ~per 20kgU | 1x / day ~per 3kgU | 10x / year ~per 100kgU | 1x / week ~per 20kgU | 1x / day ~per 3kgU |
| **Fission Xe in g:** | 154 | 31 | 5 | 12 | 2 | 0.4 |
| **Fisson Xe : atm. Xe after 1 h:** | 1 : 0.3 | 1 : 1.4 | 1 : 9 | 1 : 4 | 1 : 18 | 1 : 120 |

**Table 1:** Xenon fissiogenic inventory and signal to noise ratio

The expected signal to noise ratio is quite high for the proliferation sub-case of 10 dissolutions per year compared to the daily proliferation sub-case.

## 4.2. Isotopic alterations

Once released stable noble gases mix with atmospheric noble gases. To obtain a reliable estimation of the expected changes in the isotopic composition of atmospheric xenon due to blending with released fission xenon, the “simulation of the dilution process” for scenario A, B, C and D was performed by examining the isotopic alteration of xenon at different blending ratios. These ratios correspond to an estimated air flow rate in the stack of $10^4$ m$^3$/h, $10^5$ m$^3$/h, $10^6$ m$^3$/h and to the discussed proliferation subcases for undeclared reprocessing. Figure 1 shows the change in the $n(^{136}Xe)/n(^{132}Xe)$ for a daily reprocessing activity with increasing stack dilution factor compared to atmospheric xenon.

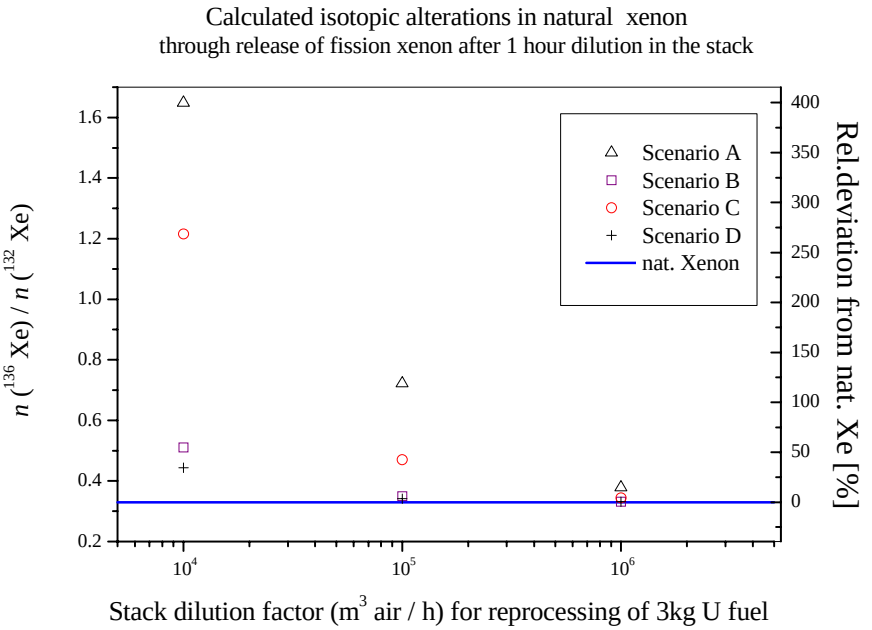



**Figure 1:** Isotopic alterations for a daily reprocessing activity

Figure 2 shows the change in the $n(^{131}Xe)/n(^{132}Xe)$ for a weekly reprocessing activity with increasing stack dilution factor compared to atmospheric xenon.

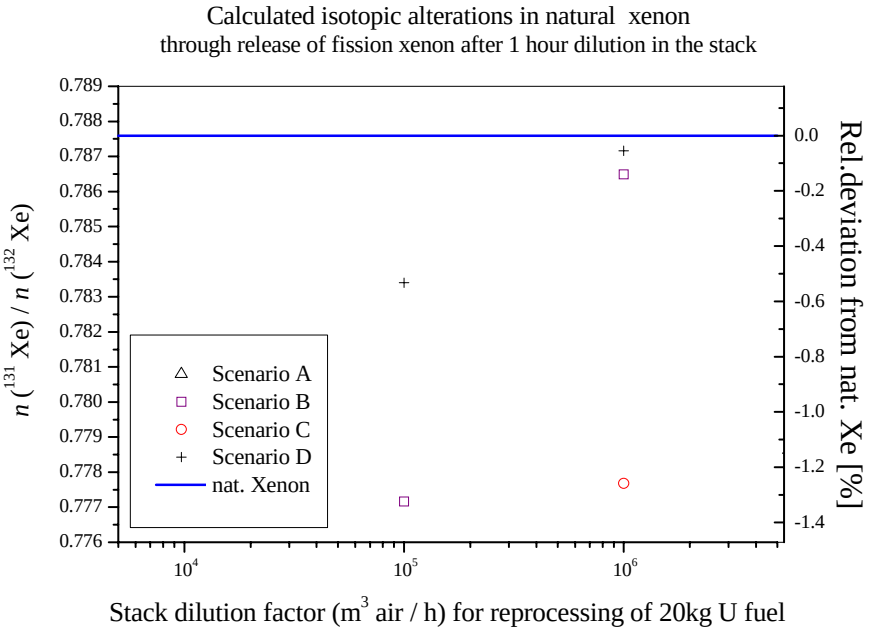



**Figure 2:** Isotopic alterations for a weekly reprocessing activity

The deviation from the atmospheric isotope ratios is significant and differs for the high and low burn-up scenarios. By measuring these kind of alterations in the blend an by measuring the isotopic composition of atmospheric xenon conclusions can be drawn on the isotopic composition of the initially released noble gases by simply applying the Isotope dilution equation with the constraint that the light isotopes are not produced via nuclear fission [2]. The calculated initial isotopic composition of

he fission off gas at the dissolver could be used as input parameter for models that render probabilities on fuel parameters and reactor operation.

## 5. Mass Spectrometry

The isotopic measurement procedure at IRMM for highly accurate gas isotope ratio measurements is based on the controlled gas flow from the mass spectrometer inlet system to the ion source through a molecular flow gold leak. Therefore the signal acquired at the detector depends exponentially on the gas partial pressure and the time. The observed ion currents are corrected for this mass discrimination and by means of gravimetrically prepared synthetic isotope mixtures for any residual errors [3, 4]. In order to measure small isotope ratios, new hardware improvement has been achieved recently by introducing a combined Secondary Electron Multiplier (Ion counting) and Faraday detection. This allowed the measurements of isotope ratios lower than $10^{-7}$ and therefore opened a large field of investigation for natural isotope studies and highly enriched isotopic samples. This progress in mass spectrometry techniques increases possibility to measure all the required mass-to-charge signals for the complete determination of the isotopic fractional abundance of noble gases.

### 5.1. Limit of detection

A set of suitable ratios to detect undeclared nuclear activities were identified as result this feasibility study The limits of detection resulting from this feasibility study assuming high accurate mass spectrometry techniques are summarised as follows:

- ***n*($^{136}$Xe)/*n* ($^{129}$Xe), *n*($^{134}$Xe)/*n* ($^{129}$Xe), *n*($^{128}$Xe)/*n* ($^{134}$Xe)**
  - Very suitable for detection of undeclared nuclear activity: Detection of alteration to a dilution factor in natural xenon of about 1:100 000; i.e. 2 - 3·$10^8$ $m^3$ air per emission of 1 g fission xenon

- ***n*($^{136}$Xe)/*n* ($^{132}$Xe), *n*($^{134}$Xe)/*n* ($^{132}$Xe), *n*($^{131}$Xe)/*n* ($^{134}$Xe)**
  - Suitable for detection of undeclared nuclear activity: Detection of alteration to a dilution factor in natural xenon of about 2 - 3·10 $m^3$ air per emission of 1 g fission xenon
  - Suitable for detection of undeclared nuclear activity within a declared facility: Distinction between scenarios to about 5·$10^6$ -2·$10^7$ $m^3$ air per emission of 1 g fission xenon
  - The detection limit for high accurate measurements after 1 hour for isotopic alteration of *n*($^{136}$Xe)/*n* ($^{132}$Xe) due to proliferation reprocessing (low-burn-up) of 3kgU - 100kgU corresponds to a stack dilution factor of 1·$10^8$ - 5·$10^9$ $m^3$ air / h

- ***n*($^{131}$Xe)/*n* ($^{132}$Xe)**
  - Very suitable for detection of undeclared nuclear activity within a declared facility: Distinction between scenarios to about 5·$10^7$ $m^3$ air per emission of 1 g fission xenon

To be complete detection limits for Kr are also given

- ***n*($^{86}$Kr)/*n* ($^{84}$Kr)**
  - The detection limit for high accurate measurements after 1 hour for isotopic alteration of *n*($^{86}$Kr)/*n* ($^{84}$Kr) due to proliferation reprocessing (low-burn-up) of 3kgU - 100kgU corresponds to a stack dilution factor of 7•$10^5$ - 2•$10^7$ $m^3$ air/h.
  - Detection of alteration to a dilution factor in atmospheric krypton of about 3•$10^7$ $m^3$ air per emission of 1 g fission xenon. Distinction between low and high burn-up to about 3 •$10^5$ $m^3$ air per emission of 1 g fission xenon

## 6. Potential of the stable noble gas monitoring

The relative measurement uncertainty on Xe and Kr isotope ratio measurements is 0.001% for high accuracy measurements and 0.1%-0.5% for routine measurements. The feasibility study has proven

that all assumed proliferation sub-cases could definitely be detected from an analytical point of view for a stack dilution factor of $10^5$ $m^3$ air / h even by means of routine stable xenon isotope ratio measurements. Noble gas monitoring definitely has a potential to detect and identify undeclared activities. The detection of the change in xenon isotopic ratios from the natural abundance would be challenging but feasible in samples taken up to 1 hour after release in the case of access to the stack. It needs to be investigated whether existing stack monitoring systems could be used with slight modifications for noble gas sampling.

Besides this application, stable xenon and krypton measurements have a high potential to other safeguards related applications. Stack sampling and measurements of stable noble gases could also be used for confirmation of operation declaration concerning the fuel that has been reprocessed at declared large reprocessing plants in addition to swipe sample analysis.

## 7. Quality control (QC)

Reliability and comparability of analytical results on stable noble gases are of course an indispensable prerequisite for IAEA safeguards application. Analysis of noble gas samples could be done using a similar network of analytical laboratories approach as for swipe sample analysis. There are laboratories working in the geological and cosmological research field that perform high quality measurements on stable noble gases already successfully for years. A laboratory-based analysis could establish a network of already existing laboratories that fulfil the IAEA quality assurance requirements that are in alignment with ISO 9000 or ISO 17025. The laboratories would need to analyse a number of QC samples and to participate regularly in recognised Interlaboratory Comparison schemes in view of quality assurance and to use high quality reference materials for their method validation. Once implemented stable noble gas monitoring could be a cost-effective tool for nuclear safeguards purposes.

## 8. Conclusions

For strengthening the IAEA safeguards system, in the verification of the completeness and correctness of a states declaration, all available characteristic signatures need to be taken into account. The information that could be revealed by stable noble gas isotope measurements had not been fully recognized so far. In the present study we demonstrated that the method has a high potential for being applied as supplementary technique. In particular, it could be shown that its strengths are in the wealth of information on activities going on inside declared reprocessing facilities. This information can be gathered in a mostly non-intrusive way, i.e. through sampling at the off-gas stack. The conclusion was reached that characteristic xenon and krypton signatures originating from irradiated fuel are a promising additional tool for confirmation of operation declaration of fuel reprocessed at large nuclear reprocessing plants. Within some constraints the accumulated information gained from xenon signatures could also provide valuable information for nuclear safeguards verification of undeclared reprocessing activities, supplementary to radiometric measurements of $^{85}Kr$. The recommendation of the technical expert group summarised in the report to the IAEA, STR-351, was that more numerical simulations need to be performed with subsequent experimental verification and a cost-benefit study.

## 9. Acknowledgements

The authors very much acknowledge the support from Paolo Peerani (JRC-IPSC) who provided the fissiogenic inventory data for this feasibility study.

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# Development of an atmospheric $^{85}$Kr automated sampler and analyzer: SPARK

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**Abstract.**

*One of the potential interests of atmospheric $^{85}$Kr measurement is the detection of clandestine reprocessing activities as 10-35 TBq of $^{85}$Kr are emitted per kg of reprocessed Pu. To this goal, a system called SPARK (French acronym for Système de prélèvement et d'Analyse automatique du Radio-Krypton) for krypton sampling, purification, concentration, and $^{85}$Kr measurement has been developed. This system is fully automated, has a 6-hour operating cycle, trap krypton at room temperature, and can detect moderate $^{85}$Kr activity variations, less than 0.1 Bq·m$^{-3}$ above background (1.45 Bq·m$^{-3}$ in the NH). In the first part, we describe the SPARK system. It is based on the use of a permeation membrane that purifies air from $O_2$, $H_2O$, $CO_2$, and pre-concentrates Krypton by a factor of three, and on room temperature trapping - high temperature desorption on high specific area active charcoal beds for purifying and concentrating krypton. Final krypton volume transferred into the measurement cell is 0.7 cm$^3$. Low-level $^{85}$Kr measurement is performed thanks to a modified proportional counter. In the second part, we present and discuss the results obtained with the SPARK system, operated for several weeks in Bruyères-le-Châtel CEA research centre. Measured $^{85}$Kr activities are compared with atmospheric transport modelling system, based on the FLEXPART Lagrangian dispersion model. It was elaborated to analyse and explain the $^{85}$Kr detections obtained by SPARK. In the context of the detection of clandestine reprocessing activities, the motivation is the spotting of the space and time location of the krypton sources. The calculated source terms required to get the detections recorded by the SPARK must be compared with the inventoried sources in order to suppress unrealistic estimations in terms of contributing facilities. Calculations show that the krypton sampler is mainly under the influence of reprocessing plants located in Western Europe.*

**Keywords**: krypton-85; reprocessing; detection; atmospheric modelling

**Foreword:** Since a few years, the IAEA has become aware of growing challenges, including the cover acquisition and clandestine operation of sensitive nuclear fuel cycle technologies able to produce nuclear material necessary to manufacture nuclear weapons, among which reprocessing techniques. This awareness has been fuelled by the announcement by North Korea that it has reprocessed spent fuel previously under AIEA monitoring and the chance of new reprocessing campaigns to occur. Iran is also developing troublesome Plutonium production capabilities. The measurement of the atmospheric concentration of $^{85}$Kr could improve the IAEA capabilities to detect undeclared reprocessing activities. This Technique proposed by France on the base of its CTBT noble gas monitoring experience, has been selected as one of the seven novel techniques and instruments for detection of undeclared nuclear facilities, material and facilities to be developed in the framework of the project SGTS SG-08[1].

[1] Research and Development Programme for 2006-2007/IAEA Department of Safeguards/January 2006

To carry out the research and development on prototype equipment, a task has been proposed to France (06/TDO-005) which is still in reviewed[2].

## 1. Introduction

Krypton-85 ($\tau_{1/2}$ = 10.76 yrs) is a fission product that exists in nuclear fuel for years after the fuel has cooled. The specific content of $^{85}Kr$ per ton of fuel varied between 50 and 340 TBq depending on the type of reactor and the operational history of the fuel rods [1]. In a similar way, the content of Plutonium in a fuel rod may vary by about an order of magnitude. In the course of the production of "weapon grade" Pu (≤ 7% $^{240}Pu$), 10 to 35 TBq of $^{85}Kr$ are released per kg of Plutonium [1]. Over the last several years, interest has arisen in the use of $^{85}Kr$ as a tool for detecting signs of clandestine reprocessing [1-4].

However, $^{85}Kr$ atmospheric background is relatively high. According to Steinkopff [5], the mean $^{85}Kr$ activity for weekly samples collected between 2001 and 2003 amounts to 1.5±0.5 $Bq \cdot m^{-3}$ for the Offenbach site (Germany), and 1.6±0.6 $Bq \cdot m^{-3}$ for the Freiburg site (Germany). Moreover, it exhibits strong variations [5-7] that are mainly due to a combination of releases from reprocessing facilities and atmospheric dilution. Although sample variability is high, the maximal value reached "only" 3.5 $Bq \cdot m^{-3}$ (in September 2001), and only a few weekly samples were between 2 and 3 Bq $m^{-3}$ [5]. Thus, even in these locations, influenced by releases from La Hague, France, and Sellafield, UK, weekly detection of $^{85}Kr$ at levels above 5 $Bq \cdot m^{-3}$ may potentially be detected which give indication on the capability to detect clandestine activity.

Moreover, the amplitude of these variations strongly depends on the geographical area and on distance from the main reprocessing facilities. Hundred of kilometres away from these facilities (for instance in the Southern Hemisphere), available measurements show that relative variations are relatively low (a few %) and close to the uncertainties of the measurements [6,7]. Then, in these areas, detection of moderate increases of the $^{85}Kr$ activity (a few tenth of $Bq \cdot m^{-3}$ above background) may be an indication of a clandestine reprocessing.

In the past decades, some laboratories developed systems for low-level atmospheric $^{85}Kr$ measurements especially for purposes of environmental survey of atmospheric radioactivity and global modeling of atmospheric transport [6-10]. However, these systems are manual, and used liquid Nitrogen to increase the efficiency of Krypton trapping on solid adsorbents. Therefore, such systems require presence of skilful staff and a well-equipped laboratory. Generally speaking, techniques for collecting and separating $^{85}Kr$ from the atmosphere are similar, however, to those described for radioactive Xenon isotopes, but are less efficient for Krypton than for Xenon. Unfortunately, no commercial system exists to collect and analyze $^{85}Kr$ at environmental concentrations.

The aim of this work is to develop an automated system for Krypton sampling, concentration, purification, and for $^{85}Kr$ measurement. This system is called SPARK (French acronym for Système de prélèvement et d'Analyse automatique du Radio-Krypton). For this work, we benefit from experience gained during development of the Spalax™ Noble Gas Equipment for CTBT implementation [11]. Indeed, some features of this equipment are similar to those of the "Spalax™" equipment.

SPARK was designed according to the following technical specifications: i) fully automated sampling, treatment (concentration and purification), and measurement; ii) trapping at ambient temperature (no cryogenic cooling); iii) a 6-hour duty cycle; iv) $^{85}Kr$ measurement thanks to a proportional counter; v) obtention of at least 0.5 $cm^3$ of Krypton. This volume theoretically provides a detection capability of a few tenth of $Bq \cdot m^{-3}$ of $^{85}Kr$ over background, assuming given performance of the proportional counter (efficiency, background) and given precision over Krypton volume determination.

[2] Summary of Decisions and Agreed Action Resulting from the French Support Programme Annual review Meeting,, Vienna, 19 June 2006.

## 2. Description of the equipment

The SPARK prototype is composed of four distinct stages that perform successively the following steps: i) sampling, first purification and pre-concentration; ii) purification from radon and concentration; iii) further concentration; iv) $^{85}$Kr activity and of Krypton volume measurements. We give a diagram of the whole system in the Figure 1 below. Homemade software based on National Instrument acquisition card allows to control and command the whole system, and to register useful data.

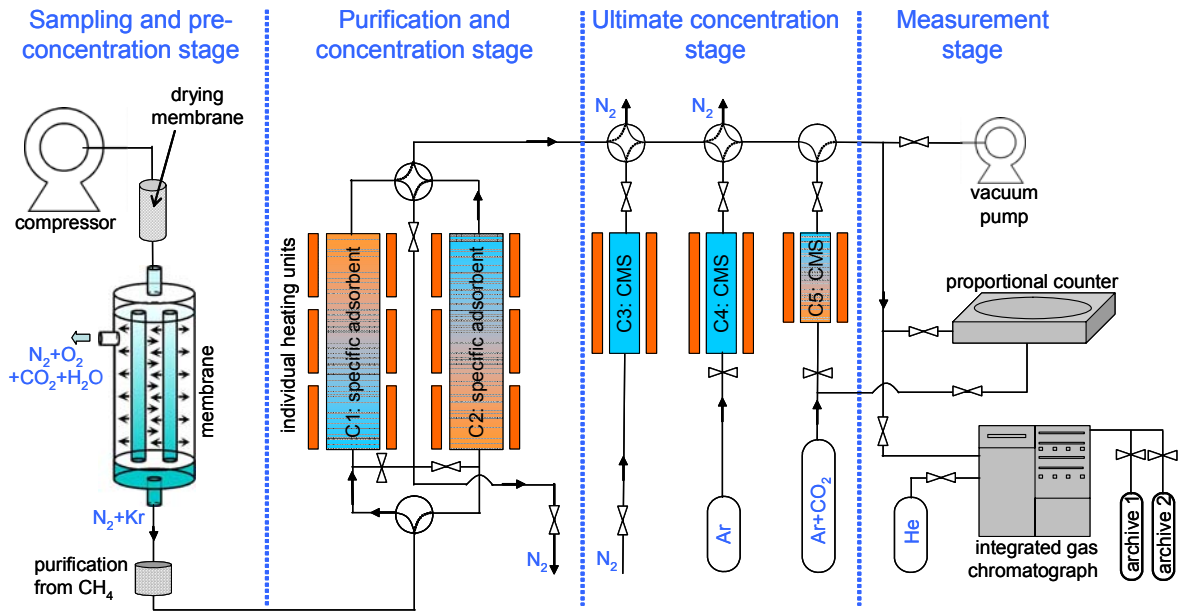


Figure 1. Diagram of the atmospheric $^{85}$Kr automated sampler and analyzer.

### 2.1. Sampling, first purification and pre-concentration stage

This first stage must achieve three simultaneous goals. The first one is to sample ambient air with a high flow rate. The second is to produce a gas mixture convenient for adsorbent like activated charcoal, i.e. to eliminate most of $O_2$ and $CO_2$ that are poisons for the adsorbent involved in the next step of the process. The third one is to pre-concentrate Krypton. We achieved all these goals by using a permeation membrane originally designed for pure Nitrogen production. We tested several commercially available membranes were tested for their capacity to enrich Nitrogen with Krypton. The most efficient one provides an enrichment factor of 2.9 i.e. the Krypton concentration, measured at the Nitrogen outlet, is close to 3.3 ppm instead of 1.14 ppm in ambient air [12]. In addition to purification and pre-concentration, this arrangement present also the advantages to be a passive (no power supply required), continuous (does not require any regeneration time) and robust device.

It is necessary to compress the sampled air under about 8 bars in order to obtain optimum performance of the membrane. Compressed air is then dried by means of a drying membrane (-40° dew point). The flow rate at the inlet of the Nitrogen generator is about 4 $m^3 \cdot h^{-1}$. In order to obtain optimum Krypton enrichment, the Nitrogen flow rate, at the outlet of the Nitrogen generator, is restricted to about 100 $L \cdot h^{-1}$. Finally, the gas mixture (mainly dry Nitrogen enriched with Krypton) is oxidized on catalytic Platinum (400°C) in order to remove traces of Methane. Therefore, air equivalent processed volume in 6 hours is about 0.6 $m^3$. Therefore, the global Krypton recovery yield $\eta$ is relatively low at optimum enrichment value. $\eta$ is given by:

$$\eta = 100 \cdot E \cdot \frac{\phi_{output}}{\phi_{input}}$$

Where E is the Krypton enrichment factor (2.9) and $\phi_{input}$ and $\phi_{output}$ are respectively the input and output flux (respectively 4 $m^3 \cdot h^{-1}$ and 0.10 $m^3 \cdot h^{-1}$). Thus, we obtain: $\eta \approx 7.25\%$ .

## 2.2. Purification and concentration stage

The second stage of the process is dedicated to Krypton trapping and concentrating and Radon removal. Room temperature continuous adsorption of Krypton is achieved by using a dual-bed system of 1m long, 30mm diameter thick Copper column (named C1, C2). Both are divided in three sub-columns (named C11, C12, C13 for the first column and C22, C21, C23 for the second column). Each of them is filled with 100 grams of "molecular sieve" carbon and located in a tubular oven. The breakthrough time of each 1m-column is fitted to be close to 1 hour, i.e. the time the oven spends to cool down to ambient temperature

Each C1 sub-column is then regenerated successively by raising the temperature up to 300°C and flushed by means of a moderate Nitrogen flux. For example, Krypton and Nitrogen mixture delivered downstream from the sub-column C11 is added (or adsorbed again) on C12 column and so on. Ultimately, the whole Krypton retained in the 1m-column is adsorbed on its last third i.e. C13 sub-column. That desorption procedure allows to extract all Krypton by means of one third of the Nitrogen volume needed to extract Krypton at one go from the 1m-column.

Radon is a naturally occurring radioactive gas in air whose excessive presence in the counter leads to a severe increase of background. As Radon desorption is more difficult and slower than Krypton desorption, most of the Radon remains in the "molecular sieve" carbon at the level of the two first heating units whereas Krypton is at the level of the third heating unit. Thus, this arrangement allows an efficient Radon / Krypton separation. To remove Krypton from the adsorbents, the system uses pure Nitrogen produced by the process itself. However, considering the poor adsorption capacity of most of the current adsorbents towards Krypton, we performed an adsorbent screening to choose the best one for Krypton at ambient temperature. We selected the adsorbent for its ability to trap Krypton at ambient temperature (20°C) and to provide the best resolution with respect to Nitrogen at ambient temperature [12].

## 2.3. Ultimate concentration stage

This stage achieves the ultimate Krypton concentration prior to gas introduction into the measurement stage. It consists in three in-line columns (named C3, C4 and C5), inserted in small-size furnaces. C3 is similar to C1 and C2, but its thickness is reduced. C4 and C5 are made of ¼" stainless steel tubing. These columns are filled with "molecular sieve" carbon. C3 elution and C4 adsorption are performed with pure Nitrogen, whereas C4 elution and C5 adsorption are performed with Argon. This allows to decrease as much as possible the Nitrogen concentration in the final mixture, whose composition must be compatible with the functioning of the proportional counter, i.e. Nitrogen content as low as possible. These two elution/adsorption steps allow an additional reduction of the elution gas volume by a factor of about 50. Final transfer from C5 into measurement cell is carried out without use of elution gas. The heated C5 column is directly connected to the air evacuated β-measurement cell. Desorbed gases expanded naturally into it. Then, C5 column is flushed with Ar+$CO_2$ (90% Ar – 10% $CO_2$) to transfer all Krypton in the measurement cell. First tests show that about 0.7 $cm^3$ of stable Krypton is collected (STP conditions) after thermal desorption of the last column. As the total volume of the measurement cell of the counter is 245 $cm^3$, the Krypton final concentration is about 0.3%. Therefore, the Krypton concentration factor for the whole process is about 2,500.

## 2.4. Measurement stage

Some authors report the use of liquid scintillation counting [13] and gamma spectrometry [14] for $^{85}Kr$ activity measurement. However, gamma spectrometry is not sensitive enough for low-level $^{85}Kr$ measurements and it will be a hard task to integrate liquid scintillation counting into an automated system. Proportional counting has the potential for low-level measurements of β-emitters in solid samples and many laboratories use them routinely. Nevertheless, we do not find any commercial "off-the-shelf" proportional counter that fits our needs for activity measurements of gas isotopes. Therefore, in close cooperation with a manufacturer (Canberra, Loches, France) we alter a "Mini 20" commercial proportional counter to adapt it to low-level measurement of $^{85}Kr$. The principle of this

measurement is to add the Kr produced by the previous stages of the equipment to the gas normally used to fill the counter (see Figure 2 below). Thus, gas mixture obtained after desorption of the C5 column (mainly Ar, $N_2$ and Kr) is mixed with a larger volume of 90% Ar + 10% $CO_2$. Then, this gas mixture is introduced into the counter. The counter operates in the static mode (no gas circulation) under a pressure of 1020-1060 mbars. The total volume of the chamber is 245 $cm^3$. The usual Mylar window supporting the solid state source is replaced by a Copper plate with O-ring seals to ensure airtightness. We added Copper blocks in which airtightness connections with O-ring seals can be properly fitted. Extremities of the anode wire are embedded into cavities filled with resin. Guard counters are located above and below the proportional counter containing the sample. The guard counters and the main counter are located inside a lead shield. The whole system is equipped with anticoincidence electronics. Approximate proportions of the different gases that constitute the final mixture are 87.3% Ar, 9.7% $CO_2$, 3% $N_2$, 0.3% Kr, and small amounts of $O_2$. We optimized these proportions with respect to efficiency and background. Background is about 5 counts $minute^{-1}$. Counting efficiency of this device with the aforementioned gas mixture is about 70%. Before starting, the counter is vented with a 90% Ar -10% $CO_2$ mixture, to eliminate any memory effect. At present, the altered "Mini20" counter suffers from a problem of leakage, which prevents from obtaining a stable efficiency for 6 hour - counting times, and is not yet available for routine measurements.

Krypton volume measurement is performed by a gas chromatograph ("Peri-1200", Perichrom, Saulx-le-Chartreux, France) integrated into the system. These measurements are not hindered by the presence of $CH_4$ that was eliminated before passing through the permeation membrane. Calibration of the gas chromatograph is performed thanks to three successive measurements of a gas standard with certified Kr concentration.

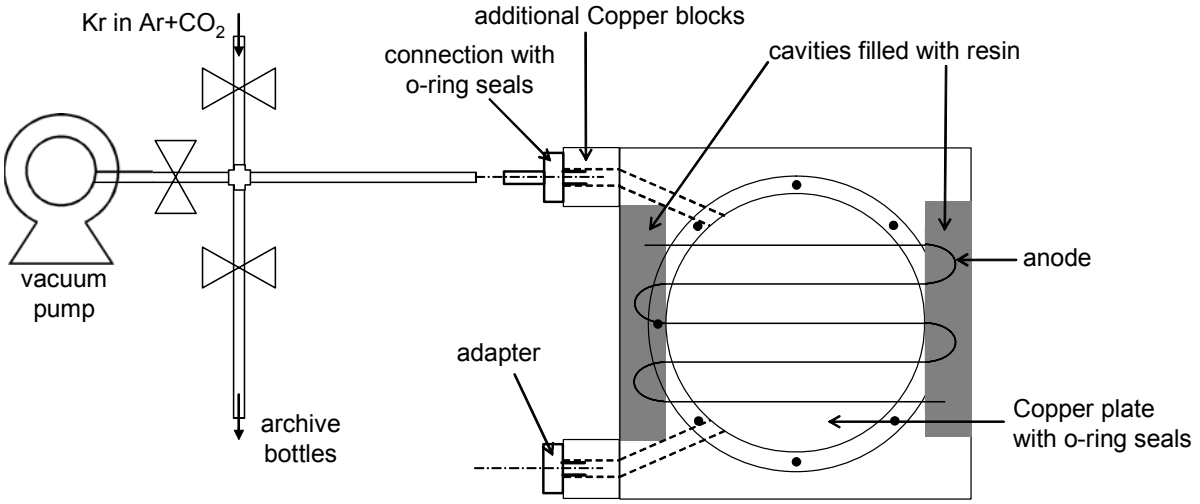


Figure 2. Diagram of the modified proportional counter. The Mylar film supporting the solid state source is replaced by a Copper plate with O-ring seals to ensure airtightness. The proportional counter is filled with the Kr and Ar+$CO_2$ mixture and operated in the static mode (no gas circulation).

## 3. System performance features

We tested the SPARK prototype, except the proportional counter, for several months. Most of the measurement cycles were done continuously to test connection of cycles. Krypton volumes sampled by the SPARK prototype from the 4$^{th}$ of October to the 26$^{th}$ of November 2006 are gathered in the Figure 3 below. Mean Kr volume obtained for 6-hour cycles is 0.71 $cm^3$. Variations of the Kr volumes are due to variations of the temperature inside the laboratory.

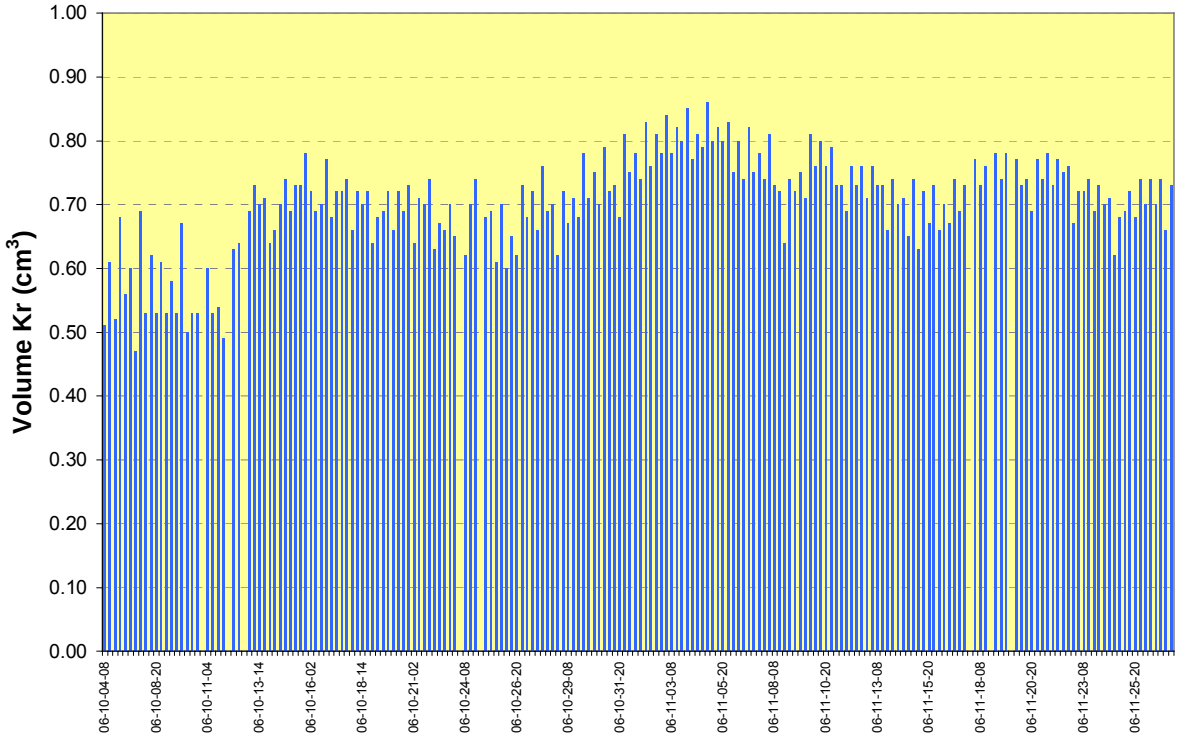


Figure 3. Krypton volumes (left axis) obtained for 6-hour cycles by the SPARK prototype, except proportional counter, from the 4th of October 2006 to the 26th of November 2006. Relative uncertainties are about 1% (coverage factor of two).

## 4. Atmospheric transport modelling

Twelve samples were sent to the Bundesamt für Strahlenschutz (BfS, Freiburg-im-Breisgau, Germany) for Kr concentration and $^{85}$Kr activity measurements. It should be noted that we have no idea of short-term (6 hours) variations of the $^{85}$Kr background in Western Europe as measurements carried out by other laboratories and described in the literature are always weekly or monthly averaged measurements [6-7].

Measured $^{85}$Kr activities are compared with activities calculated from atmospheric transport modelling system. In the context of the detection of clandestine reprocessing activities, the aim is the spotting of the space and time location of the krypton sources. At first, we need to make an inventory of the possible $^{85}$Kr sources in order to suppress unrealistic estimations in terms of contributing facilities. Main $^{85}$Kr source terms are undoubtedly La Hague and Sellafield reprocessing plants, with respectively average releases of 0.8 PBq / day and 0.25 PBq / day. Calculations show that the krypton sampler located in Bruyères-le-Châtel is mainly under the influence of these two reprocessing plants. In comparison, nuclear power plants in Western Europe are only a minor source with about 0.4 GBq / day / power plant. For these calculations, we assume that the $^{85}$Kr release flow rates are continuous ones. We used meteorological data from the Global Forecast system [15] with a temporal resolution of 6 hours and a spatial resolution of 1 degree. Atmospheric transport modelling was based on the FLEXPART Lagrangian dispersion model [16-17] that allows fast response over long period of time. In the Figure 4 below, we represented the calculated $^{85}$Kr activities ($Bq \cdot m^{-3}$) at Bruyères-le-Châtel assuming continuous releases from the two aforementioned reprocessing plants, from the 1st of October to the twelve of December 2006. We observed that influence of the two RP is clearly detectable only for 33 6-hour cycles over a total of 280 6-hour cycles, as Bruyères-le-Châtel, about 30 km South of Paris, is not under the prevailing winds from La Hague and Sellafield.

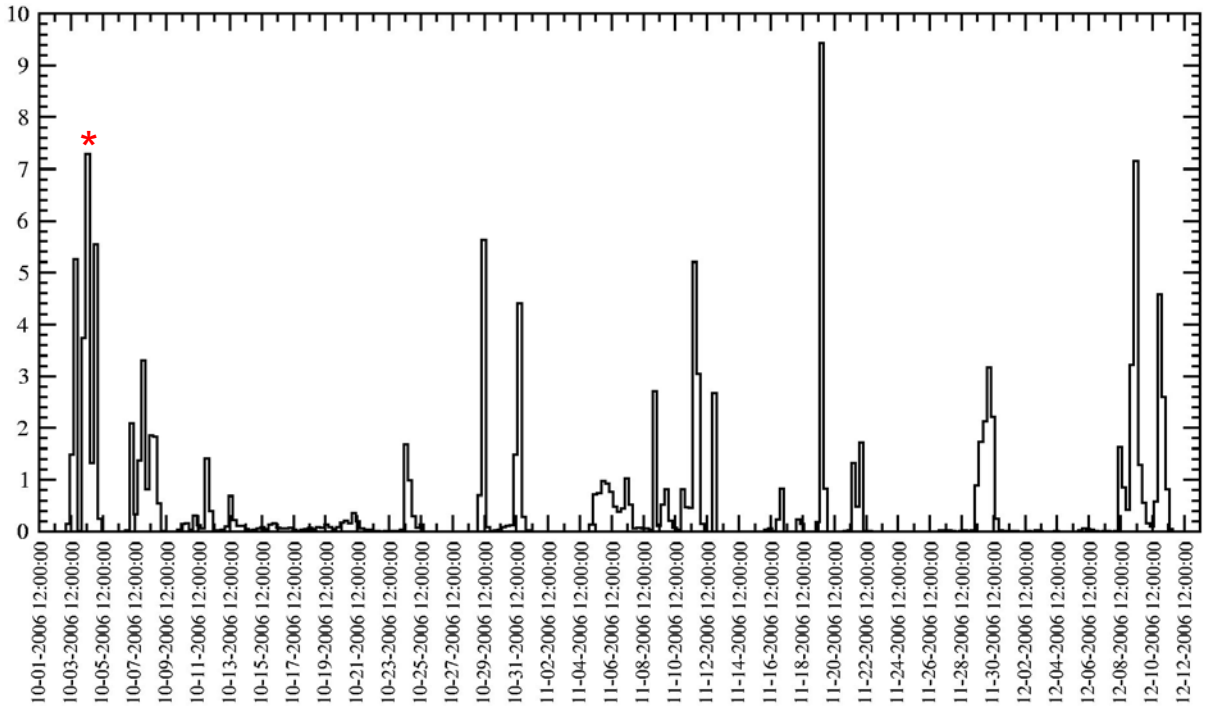


Figure 4. Calculated $^{85}$Kr activities at Bruyères-le-Châtel, where the SPARK system is presently located, assuming continuous releases from the La Hague and Sellafield reprocessing plants.

As it can be shown from the Figure 5 below, comparisons between calculated $^{85}$Kr activities using our Lagrangian code and activities measured by the BfS show a correct agreement, in spite of our over-simple hypothesis of continuous $^{85}$Kr releases. Activity measurements performed by BfS show moderate variations: from 1.5 to 3 Bq·m$^{-3}$ air.

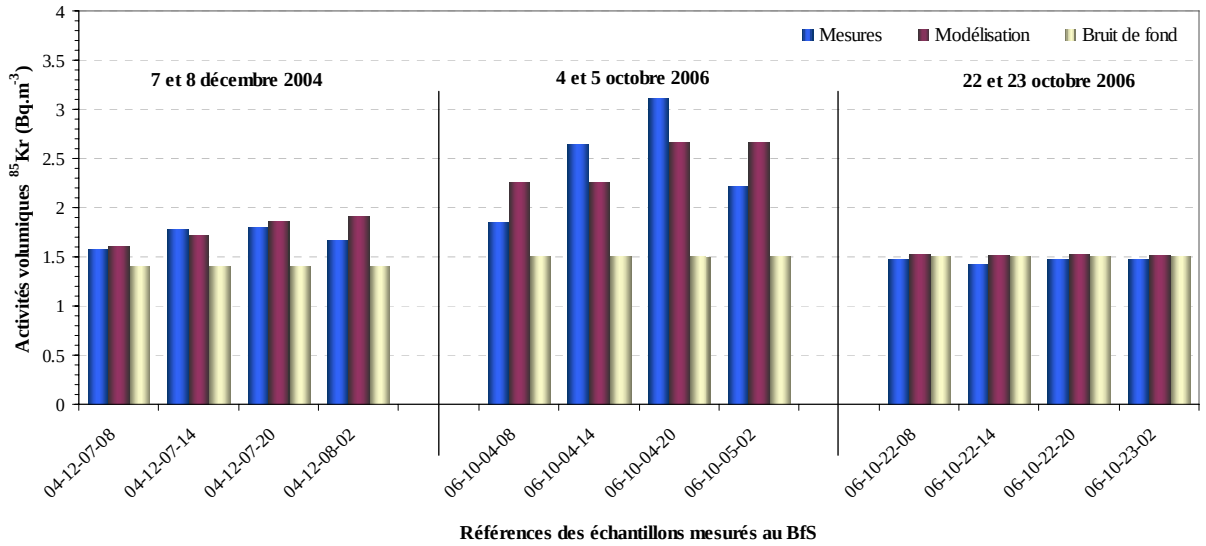


Figure 5. Comparisons between calculated $^{85}$Kr activities using our Lagrangian code and activities measured by the BfS (Freiburg-Im-Br., Germany). According to our simplistic hypothesis of $^{85}$Kr continuous releases, La Hague release flow rate is about $10^{12}$ Bq·h$^{-1}$.

## 5. Further tests and developments

Work is under way to complete the development of the proportional counter and to integrate it into the prototype system. Above all, we must solve airtightness problems that lead to slow decrease with time of the counting efficiency. Next, we have to validate the proportional counter for low-level $^{85}Kr$ measurements and integrate the counter to the sampling, purification, and concentration process. In particular, we must adapt electronics and software to start and stop automatically the counting when needed.

Afterwards, the further step will be to operate the integrated automated SPARK prototype, including this time the proportional counter, at a fixed place for 6-hour time resolution $^{85}Kr$ continuous monitoring, over a long time (at least several weeks). We will also conduct this experiment in combination with atmospheric transfer calculations. On the one hand, these measurements will hopefully provide a real improvement in the knowledge of the 6-hour resolution time variations of the $^{85}Kr$ activities near Paris, France. On the other hand, it will also allow us to evaluate the potential of such measurements for locating reprocessing facilities in Western Europe, in connection with atmospheric transfer data and information about $^{85}Kr$ releases from reprocessing plants, especially La Hague, France. In the same way, it would also be possible to state more precisely the potential of a network of Krypton automated samplers and analyzers, similar to what exist for radioxenons in the frame of CTBT implementation, for detecting and localizing illicit reprocessing activities, along with source modeling and atmospheric transfer calculations.

## 6. Conclusion

We designed and developed an automated sampling, purification, concentration, and measurement system for $^{85}Kr$. The purification from other gases, notably Radon, and the concentration are carried out thanks to a combination of gas permeation through a Nitrogen generator membrane and room temperature adsorption – high temperature desorption cycles on ”molecular sieve” carbon. The Krypton concentration factor with respect to normal air is about 2,500. The system allows to sample about 0.7 $cm^3$ of Krypton for a 6-hour operating cycle. We modified a commercial proportional counter to adapt it to low-level $^{85}Kr$ measurement. This whole system, named SPARK, has the potential for detecting automatically moderate increase, typically a few tenth of $Bq \cdot m^{-3}$, of the $^{85}Kr$ atmospheric activity above background.

We operated the whole SPARK prototype (with exception of the proportional counter for $^{85}Kr$ activity measurement) for a few months for a continuous monitoring of $^{85}Kr$ activity in the South of Paris. Comparison of activities measured by a German laboratory (BfS, Freiburg-Im-Breisgau) in samples collected by the SPARK prototype and calculated activities using a Lagrangian code show relatively good agreement, despite a simplistic assumption of continuous releases from La Hague and Sellafield reprocessing plants, the main $^{85}Kr$ emitters in Western Europe. By operating the complete SPARK system including the proportional counter for long periods of time, in combination with atmospheric modeling, we will probably learn a lot about $^{85}Kr$ activity variations with a 6 hour-time resolution and capability to relocate $^{85}Kr$ emitting sources in Western Europe. This will provide an evaluation of such automated equipment to detect and locate illicit reprocessing activities.

Along with the development of this technology, a reflexion has to be performed on possible schemes regarding the use of this technology and the needs of the IAEA, as the outcome of this process may have a feedback on the development of the system.

## 7. Acknowledgments

We are grateful to C. Schlosser and H. Sartorius from the Bundesamt für Strahlenschutz (BfS, Freiburg-im-B., Germany) for $^{85}Kr$ activity and Kr volume measurements and to D. Rivière from Canberra (Loches, France) for help and cooperation in developing a proportional counter for $^{85}Kr$ measurement.

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[17] http://zardoz.nilu.no/~andreas/flextra+flexpart.html

# Session 3

# Proliferation Resistance - I

# *Simplified Approach for Proliferation Resistance Assessment of nuclear systems*
# *SAPRA*

***Caroline JORANT***
*AREVA – International & Marketing*
*Director for Non Proliferation and International Institutions*

**ESARDA Symposium**
**Aix en Provence – May 22-24, 2007**

## *Contents*

## The general framework of the study

- A French working group has been created 4 years ago in order to address PRPP issues (« GTR3P »)
- French nuclear actors members of this group : CEA (International Department, NE Division, Military applications Division), AREVA (Corporate, NP, NC), EDF, Ministries (foreign affairs, industry), IRSN
- Purpose of this group :
  - Exchange information on worldwide activities and review publications
  - Review and analyse foreign PRPP studies
  - Develop a methodology
  - Carry out case studies (using this methodology)
- Report on methodology and case studies

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## The methodology

- Based on TOPS methodology
- Country profile
  - NNWS/NWS, technological level, type of reactors in operation, fuel cycle characteristics
- Four phases in the proliferation process
  - Diversion, Transport, Transformation, weapon fabrication, (extension of the TOPS method limited to diversion phase)
- Four barriers
  - Material barriers, Technical barriers, Institutional barriers, Specific barriers for weapon making step (extension of TOPS method limited to first three barriers)

## The methodology

- Each fuel cycle step is being considered :
  - Front end
  - Reactor
  - Back end
- A scale of values is being defined for PR level :
  - Very weak barrier (0) to very high resistant barrier (4)
  - Each individual fuel cycle step gets a mark normalized to 1 for each set of barrier (material, institutional, …) and for each phase (diversion, transport,..)
  - Marks are given by « experts » for each square of the matrix (between 500 and 1000 marks for a given scenario)



## Threats considered in this study

- Two main sets of cases :
  - Theft of nuclear materials by a Proliferant State in a foreign country
  - Covert diversion of nuclear materials in the PS own facilities
- Case studies selected just as typical examples of some current situations
  - Theft :
    - Case 1 : PWRs / closed cycle
    - Case 2 : PWRs open cycle
    - Case 3 : CANDU reactors (open cycle)
    - Case 4 : DUPIC cycle (combining PWRs and CANDUs)
  - Covert divesrsion :
    - Case 5 : PWRs only (no fuel cycle facilities)
    - Case 6 : PWRs with enrichment and fuel fabrication plants (only)
    - Case 7: PWRs with reprocessing and fuel fabriaction plants (only)
    - Case 8 : HTRs / open cycle

NOTES : 1 – In SAPRA, included in Mass and Bulk
2 – Play a role only when enrichment is needed
3 – In SAPRA, implicitely included in other technical barriers

| | | | TOPS | SAPRA | | | | Note |
|---|---|---|---|---|---|---|---|---|
| | | | | Diversion | Transport | Transform. | Weapon fabricat. | |
| MATERIAL | Isotopic | Critical mass | | | | | | 1 |
| | | Isotopic enrichment | | | | | | 2 |
| | | Spontaneous neutron generation | | | | | | |
| | | Heat generation rate | | | | | | |
| | | Radiation (of the direct use material) | | | | | | |
| | Dangerousness (=harmfullness other than irradiation) | | | | | | | |
| | Chemical | | | | | | | |
| | Radiological (other than the one of the material itself) | | | | | | | |
| | Mass and bulk | | | | | | | |
| | Physical form | | | | | | | |
| | Detectability | | | | | | | |
| TECHNICAL | Facility unattraciveness | | | | | | | 3 |
| | Facility accessibility | | | | | | | |
| | Available mass | | | | | | | |
| | Diversion detectability | | | | | | | |
| | Skill, expertise, knowledge | | | | | | | |
| | Time | | | | | | | |
| | Technical difficulty | | | | | | | |
| | Collusion level | | | | | | | |
| | Construction detectability | | | | | | | |
| | Signature of installation | | | | | | | |
| EXTRINSIC | Safeguards | | | | | | | |
| | Access / control / security | | | | | | | |
| | Location / distance (for transport phase) | | | | | | | |

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## Example of results

▶ CASE STUDY N° 1 :

**Theft in a country having a fleet of PWRs and implementing a full CLOSED fuel cycle (including a MOX fuel fabrication plant)**

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## Example of table : closed cycle – Diversion phase (theft)

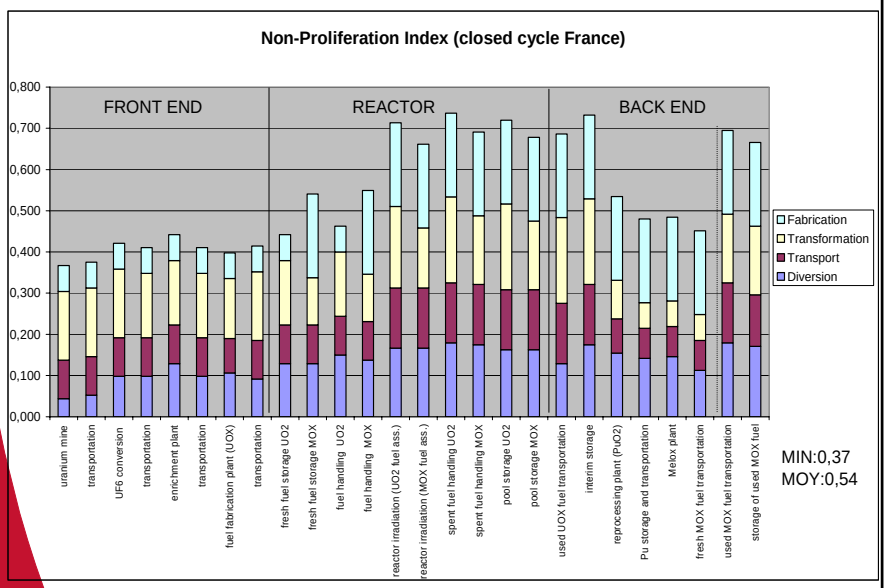
| Threat | Stages towards weapon manufacturing | diversion phase | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | barriers | material barriers | | | | | | technical barriers | | | | | |
| | attributes | radiological | dangerosité | physical form | mass/bulk | detectability | total | facility accessibility | diversion detectability | technical difficulty | collusion level | available mass | to |
| Stage of the fuel cycle where diversion occurs | | | | | | | | | | | | | |
| **Front end** | | | | | | | | | | | | | |
| uranium mine (non domestic) | | 0 | 0 | 0 | 4 | 1 | 0,25 | 1 | 0 | 1 | 1 | 0 | 0,3 |
| transportation | | 0 | 0 | 0 | 4 | 1 | 0,25 | 1 | 2 | 1 | 1 | 0 | 0,2 |
| UF6 conversion | | 0 | 2 | 2 | 4 | 1 | 0,45 | 2 | 2 | 1 | 2 | 0 | 0,3 |
| transportation | | 0 | 2 | 2 | 4 | 1 | 0,45 | 1 | 3 | 1 | 1 | 1 | 0,3 |
| enrichment plant | | 0 | 2 | 2 | 3 | 1 | 0,4 | 3 | 3 | 3 | 3 | 1 | 0,6 |
| transportation | | 0 | 2 | 2 | 3 | 1 | 0,4 | 2 | 3 | 1 | 1 | 1 | 0,4 |
| fuel fabrication plant (UOX) | | 0 | 0 | 0 | 3 | 1 | 0,2 | 3 | 2 | 2 | 2 | 0 | 0,4 |
| transportation | | 0 | 0 | 0 | 3 | 1 | 0,2 | 2 | 2 | 1 | 2 | 1 | 0,4 |
| **Reactor** | | | | | | | | | | | | | |
| fresh fuel storage UO2 | | 0 | 0 | 0 | 2 | 1 | 0,15 | 4 | 3 | 2 | 3 | 1 | 0,6 |
| fuel handling UO2 | | 0 | 0 | 0 | 3 | 1 | 0,2 | 4 | 4 | 3 | 4 | 2 | 0,8 |
| reactor irradiation (UO2 fuel ass.) | | 4 | 0 | 0 | 1 | 3 | 0,4 | 4 | 4 | insurmontable | 4 | 1 | 1,0 |
| spent fuel handling UO2 | | 4 | 0 | 0 | 3 | 4 | 0,55 | 4 | 4 | 3 | 4 | 2 | 0,8 |
| pool or dry storage UO2 | | 4 | 0 | 0 | 1 | 3 | 0,4 | 4 | 4 | 3 | 4 | 1 | 0,8 |
| **Back-end (disposal site)** | | | | | | | | | | | | | |
| spent fuel transport. to disposal | | 4 | 0 | 0 | 1 | 3 | 0,4 | 3 | 4 | 3 | 3 | 0 | 0,6 |
| handling/conditioning for storage | | 4 | 0 | 0 | 1 | 3 | 0,4 | 4 | 4 | 3 | 4 | 0 | 0,7 |
| spent fuel dry storage | | 4 | 0 | 0 | 1 | 3 | 0,4 | 4 | 4 | 3 | 4 | 0 | 0,7 |
| conditioning for disposal | | 4 | 0 | 0 | 1 | 3 | 0,4 | 4 | 4 | 3 | 4 | 0 | 0,7 |
| retreivable disposal in repository | | 2 | 0 | 0 | 0 | 2 | 0,2 | 3 | 3 | 3 | 4 | 0 | 0,6 |
| final disposal in repository | | 1 | 0 | 0 | 0 | 1 | 0,1 | 4 | 0 | 4 | 0 | 0 | 0,4 |

Barrier:
Very low = 0 | High = 3 | all the "total" values are normed to 1
Low = 1 | Very high = 4
Moderate = 2 | insurmontable

TABLE: 2 - a

EXAMPLE





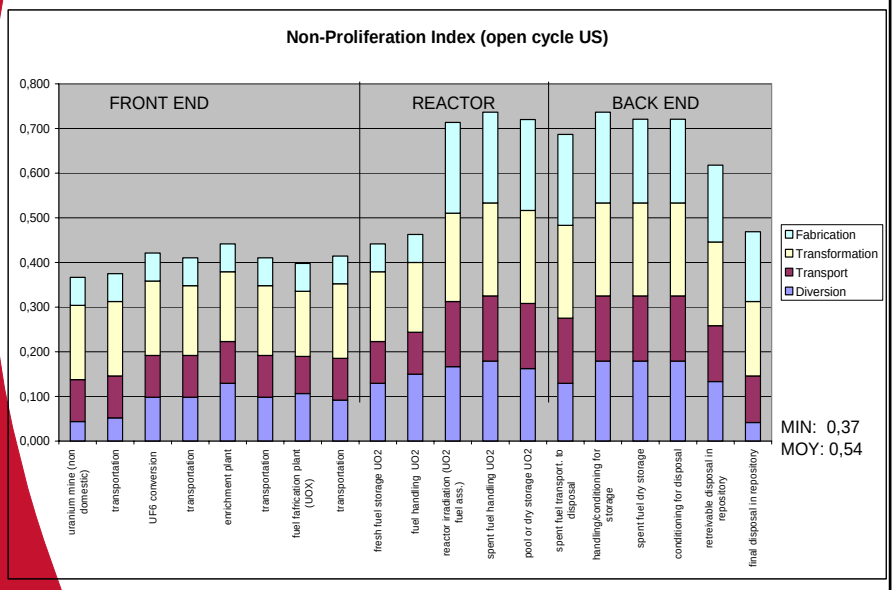

EXAMPLE

## Example of results

**CASE STUDY N° 2 :**

**Theft in a country having a fleet of PWRs and implementing an open cycle**



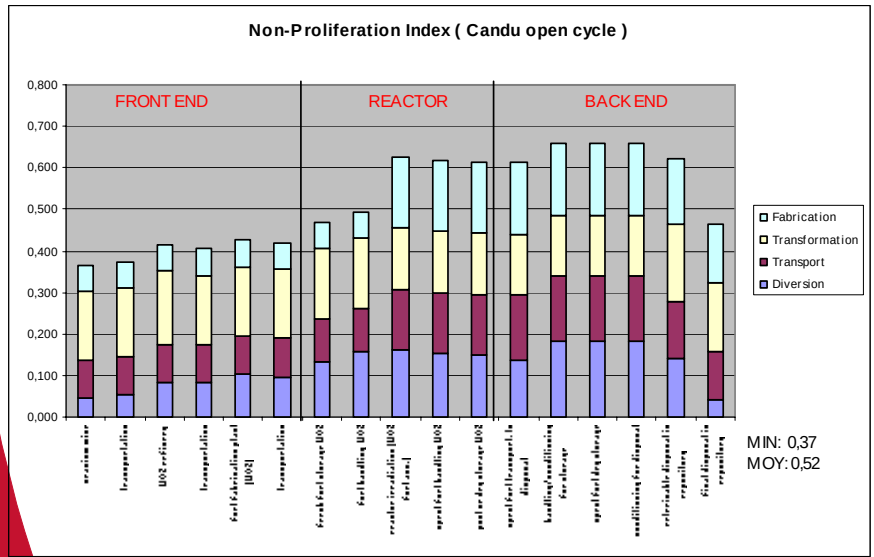


## Example of results

**CASE STUDY N° 3 :**

**Theft in a country having a fleet of CANDU reactors and implementing an OPEN fuel cycle (with a final disposal of spent fuel )**



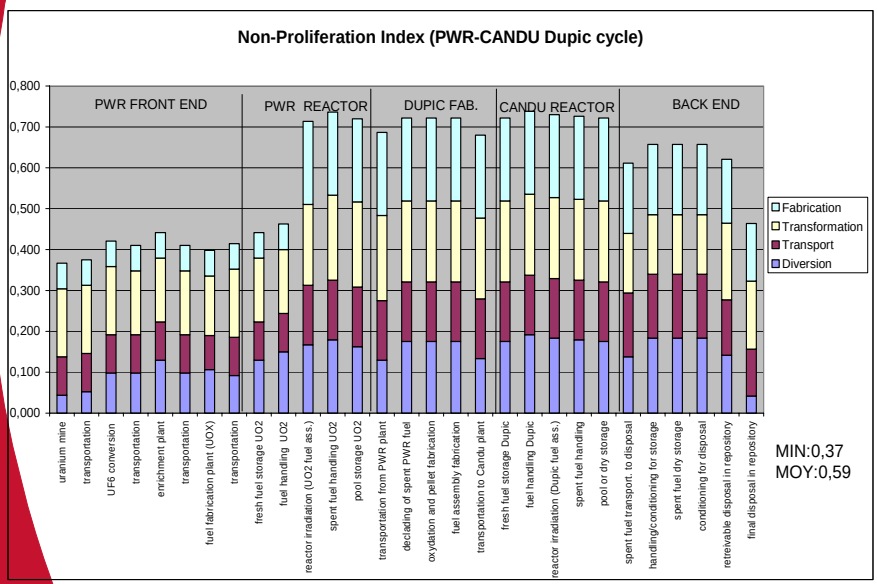


## Example of results

**CASE STUDY N° 4 :**

**Theft in a country having a fleet of PWR and CANDU reactors and implementing a DUPIC fuel cycle (with a final disposal of spent fuel )**





## Findings on Case Studies

- Few divergent assessments between experts for the allocation of marks
- The list of barriers must be carefully set up and well defined to avoid possible misunderstanding between experts and imbalance of markings
- For all cases the front end of the fuel cycle gets a lower average score than the back end
- Choice of the nuclear material liable to be used for proliferation purposes (HEU or civil plutonium) is an important feature that must be included in any PR assessment



## Conclusion on the Methodology

- The methodology is designed to assess the PR of a nuclear system as a whole (reactor and fuel cycle facilities) in the context of a specific country profile
  - Comprehensive assessment from mining to weaponization
  - Not designed to assess PR of specific facility / technology
  - Not designed to assess PP
  - Allows to detect « weak » points in the PR

***Report available upon request to D.Greneche (dominique.greneche@areva.com)***

**GTR3P[(1)] report – Revision 1** **February 2007**

# Simplified Approach for Proliferation Resistance Assessment of nuclear systems

# S A P R A

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_____

**(1) Work performed in the frame of the French Working Group on Proliferation Resistance and Physical Protection (called in French "GTR3P"), which include representatives from ministries (foreign affairs and industry), French Safety institute (IRSN), CEA (DRI, DCS, DEN, DAM), EDF and AREVA**

# EXECUTIVE SUMMARY

The aim of this report is to present a simplified method for assessing the resistance of various civil nuclear systems to nuclear weapons proliferation. Here, proliferation resistance is taken to mean that characteristic of a nuclear energy system that impedes the diversion or undeclared production of nuclear material (NM) or misuse of technologies, by States in order to acquire nuclear weapons or other nuclear explosive devices.

The application of this method, called **SAPRA** (Simplified Approach for Proliferation Resistance Assessment of nuclear systems) is illustrated by studying various cases representing typical nuclear systems currently in use today or expected to be deployed in the near future. These studies are carried out by taking into consideration various threats involving the theft or diversion of nuclear materials or the misuse of civil facilities placed under international controls. It is important to stress the theoretical nature of such studies, since any nuclear weapons arsenal built in the past has been done so without resorting to the direct use of safeguarded facilities from the civil nuclear fuel cycle.

In short, SAPRA adopts the classical multiple barriers analysis method, developed a few years ago as part of the US TOPS task force (Technological Opportunities to Increase the Proliferation Resistance of Global Civilian Nuclear Power Systems). This involves identifying a set of potential impediments or “barriers” that a proliferant state would have to overcome to reach its goals of fabricating nuclear weapons, covertly or overtly. These barriers constitute a series of obstacles that make sourcing weapon-usable materials from civilian nuclear energy activities much more difficult and/or less attractive, with minimum opportunities for diversion.

Barriers may be material (such as isotopic composition of nuclear material), technical (such as diversion detectability) or institutional (such as safeguard provisions). Unlike other approaches, SAPRA makes allowance for all barriers across the entire proliferation route that a proliferant state must follow to achieve its objectives. With SAPRA, this route, i.e. sequence of actions directed against targets, is sub-divided into four phases: diversion, transportation, transformation and nuclear weapons fabrication.

The robustness of each of these barriers is assessed throughout the main steps of the nuclear fuel cycle, from uranium mining to the final disposal of waste, since the whole civil nuclear fuel cycle involves materials that either are or could potentially be processed into weapon-usable material. For this, a scale from 0 (very low resistant barrier) to 4 (very high resistant barrier) is used. Assessments are performed on various cases, characterized by a country profile (level of technological development), a type of nuclear system (type of reactor, type of nuclear fuel cycle facilities) and a type of material acquisition (theft from a foreign installation or covert diversion of national facilities).

Once the marks are determined by a panel of experts for all barriers and all fuel cycle steps, they are then added together to give us aggregate, values normalized to 1 called “Proliferation Resistance Indexes”, or PRI. On this basis, the respective merits and weaknesses of various nuclear systems faced with various threats can be identified and discussed.

It is important to stress that this kind of simple approach cannot be used to make general comparisons between different systems in terms of their overall proliferation resistance performance. The main use of PRIs is to identify potential weaknesses and then to assess these weaknesses against extrinsic measures and against characteristics of the countries where nuclear systems are supposed to be deployed. PRIs can also be used in some cases as rough indicators that provide us with relative comparisons for selected fuel cycle steps.

On the basis of this study, several conclusions are drawn, either on the method itself or on the results obtained by applying this method.

Certain improvements to the method that could be introduced in a future development are also proposed. It is suggested in particular that such developments could benefit from the formulation proposed in the so called “Multi-Attribute Utility Analysis” developed in particular by the Texas A&M University in the USA.

As for the results, this study shows that all nuclear systems generally have a relatively high resistance to proliferation, provided that comprehensive and efficient international controls are implemented. Nevertheless, in most cases, it appears that the front end of the nuclear fuel cycle is fairly less resistant than the rest of the nuclear fuel cycle. Uranium mines in particular seem to be a weak point, since diverting natural uranium appears easier than diverting low enriched uranium in highly protected nuclear facilities. This would not make a very large difference to a potential proliferant state, which would in any event need an enrichment capacity. This conclusion remains valid even for a closed cycle involving plutonium handling, because of the very high level of protection measures specifically implemented in that case, and because of enhanced degradation of the plutonium quality when plutonium is recycled. Moreover, this fuel cycle option avoids the final disposal of used fuels in a geological repository. As a matter of fact this could raise proliferation concerns in the long term (after the final closure of the repository), because used fuels contain large amounts of plutonium which are less and less protected by the radiological barrier, and could therefore constitute “plutonium mines”.

Additional results are also discussed in the conclusion, principally with regard to HTRs and the DUPIC fuel cycle. In particular, we see that the DUPIC cycle clearly presents advantages in terms of proliferation resistance in the back end of the nuclear fuel cycle. Also, while HTRs have a weaker front end when compared with PWRs (mainly because of their use of a significantly higher enrichment of uranium), the opposite is true of their back end, because of the very nature of this system’s fuel (high burnup, high dilution of the plutonium in the fuel, difficult to reprocess, etc).

# "Generation IV" Safeguards Instrumentation

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## *Abstract*

In the frame of the Generation IV International Forum (GIF), the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO), and, more recently, the Global Nuclear Energy Partnership (GNEP), experts discuss the future use of nuclear power by addressing a variety of areas, ranging from new nuclear reactor technologies to international fuel cycle models. Aside from economic and inherent safety issues, considerations on proliferation resistance have gained increased international attention, building an important foundation for the feasibility of nuclear fuel supply and fuel cycle services models. Proliferation resistance is ruled by both intrinsic and extrinsic factors. Intrinsic factors are related to the quantities and quality of nuclear materials used in any given nuclear facility and the ease with which both materials and technologies could be withdrawn from the installation. Extrinsic features stem from institutional barriers against diversion or misuse and relate mainly to the application of International Atomic Energy Agency (IAEA) safeguards. Looking forward towards future nuclear technologies, the question arises how these will impact the future safeguards culture and supporting instrumentation. The following paper will address this question and discuss some future aspects of safeguards by extrapolating and expanding on the evolution of safeguards from a material and technology based control system to an information-driven approach. Furthermore, factors will be outlined that may impact not only the development and implementation of future safeguards instrumentation, but also the design of future nuclear reactors. Features of safeguards instrumentation may be ranging from remote interrogation capabilities to multipurpose, synergy-enabling functions, i.e., the consequences of an expected increase in a global nuclear market within a 'nuclear renaissance' on future safeguards instrumentation will be highlighted. Also, the need for early involvement of all concerned parties, especially treaty verification authorities, will be discussed. Considerations on how the non-proliferation community can best become prepared for the technological needs of the future will conclude the paper.

**Keywords**: proliferation resistance, safeguards, technologies, nuclear renaissance

## Introduction

In recent years, the civil use of nuclear energy has become an increasingly discussed topic. Especially rising prices for oil and other fossil fuels and concerns about climate change effects prompt more and more countries to revisit their nuclear programs or to investigate the addition of emission-free nuclear electricity generation to their energy portfolio. At the same time, geopolitical events such as the nuclear weapons test conducted by the Democratic People's Republic of Korea (DPRK) or the much disputed nuclear fuel enrichment program in Iran have raised new concerns about the proliferation of sensitive nuclear technologies and materials and the subsequent covert development of nuclear weapons in more and more states.

A variety of international initiatives, discussion groups, and topical workshops aims at alleviating such concerns by offering alternative approaches to the use of nuclear energy that have built-in proliferation resistance features while also addressing other important issues such as economic profitability or safety. Experts have been discussing the implementation of proliferation resistance features in new nuclear installations and fourth generation nuclear power plants in the Generation IV International Forum (GIF) or the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO). More recently, the concept of multi-national fuel cycle models has manifested in the Global Nuclear Energy Partnership (GNEP) where sensitive nuclear technologies are restricted to some countries,

whereas guaranteed fuel supply and waste management services allow any participating country to build and operate nuclear power plants without sustaining their own full fuel cycles.

Looking forward to the implementation of both new reactor types and multi-national fuel cycle models the question arises how this will impact the future international safeguards and treaty verification culture as well as the instrumentation that will support it. There will certainly be challenges to the safeguards regime but also opportunities, as new instrumentation offers not only new and enhanced verification capabilities but also synergies with other areas critical to the success of future nuclear reactors or institutional models.

The following paper will outline the basic principles of proliferation resistance, the impact of intrinsic and extrinsic factors that drive it, and their importance within a changing safeguards culture that moves from a material and technology-based control system to an information-driven approach. The paper will then discuss the role of instrumentation in the support of new standards not only for safeguards implementation but also for security, physical protection, or personnel safety. The impact on the design of future nuclear installations is discussed next. Lastly, the need for early involvement of concerned parties, especially of treaty verification authorities, will be discussed along with considerations on how the non-proliferation community can drive the process and become best prepared for the technological challenges of the future.

## Proliferation Resistance

In the context of assessing the capabilities of civil nuclear fuel cycles to resist the intentional misuse for undeclared nuclear weapons programs, *Proliferation Resistance* can be defined as "that characteristic of a nuclear system that impedes the diversion or undeclared production of nuclear material, or misuse of technology, by States in order to acquire nuclear weapons or other nuclear explosive devices."[1] Generally, two groups of proliferation resistance factors can be distinguished: intrinsic and extrinsic features. The same group of experts that met in Como, Italy, during October 28 – 31, 2002, and that defined proliferation resistance as quoted above agreed on the following definition of intrinsic and extrinsic features. 'Intrinsic proliferation resistance features are those features that result from the technical design of nuclear energy systems, including those that facilitate the implementation of extrinsic measures. […] Extrinsic proliferation resistance measures are those measures that result from States' decisions and undertakings related to nuclear energy systems.'[2]

Intrinsic features comprise the implementation of technical characteristics to the nuclear fuel cycle that reduce the attractiveness of the generated materials for diversion. These features can be applied to all steps of the nuclear energy system: production, use, transport, storage, and disposal. Other intrinsic features are those that prevent diversion or undeclared production, reduce the number of diversion paths, and facilitate continuous verification of the present nuclear materials inventory.

Extrinsic measures, on the other hand, include features such as administrative decisions on the implementation of intrinsic features early in the design and development of the nuclear fuel cycle, verification and control agreements on national and international levels, defense in depth through complementary and redundant proliferation resistance features, and institutional and legal agreements to control the access to nuclear materials and technologies.

Certainly, the extrinsic and intrinsic features that can be imagined vary both in the strength they add to the proliferation resistance of a nuclear reactor or a broader fuel cycle model and in the difficulty and cost with which they could be implemented. The development of a proliferation resistance assessment methodology as it was envisioned by the Como group to analyze different measures both from a quantitative and qualitative view turned out to be very difficult to formulate due to the difference in character of extrinsic and intrinsic features.

Institutional measures and international agreements draw their strength from the combined force that the international community can muster in case an adversary is identified. They are, however, dependent on a credible treaty compliance verification regime that can detect diversion with an acceptable probability. Intrinsic measures such as the use of nuclear materials unattractive for weapons programs or the design of reactors that make materials extraction extremely difficult can be implemented during the design and construction of new nuclear installations. But such are static

features that can neither adapt to new challenges to the non-proliferation community nor are they cheap.

Balancing extrinsic and intrinsic features that are so inherently different from each other on the level for which they provide proliferation resistance is extremely challenging both from a value and a cost point of view. The problem is further complicated by the fact that proliferation resistance assessment can not be conducted without taking into account other factors such as physical protection, safety, environmental concerns, and especially economical profitability and sustainability. \ A nuclear reactor or international fuel cycle model with high proliferation resistance rating is worthless if it is not affordable.

Even though it can be envisioned that certain extrinsic and intrinsic measures are synergetic with other factors, there are others that are antagonistic, requiring a priority decision. Bearing this multi-dimensional problem in mind, it seems that the creation of a proliferation resistance assessment methodology might be possible solely for the factor of proliferation resistance, but impossible in the practical sense when analyzing the whole spectrum of factors. Even the call for expert judgment to replace the implementation of value assessment seems problematic, as it will introduce a subjective element that might be unconsciously impacted by factors other than proliferation resistance.

However, the complexity of the decision process should not prevent the selection of an appropriate mix of extrinsic and intrinsic proliferation resistance features. A proliferation resistance checklist could be used to ensure that all areas and matters related to proliferation resistance are addressed during the design stage of a nuclear power reactor or a multi-national approach. A sample checklist can be found in a report published by the U.S. Department of Energy, Nuclear Energy Research Advisory Committee (NERAC). It comprises material, technical, and institutional barriers to proliferation is identified.[3]

In direct comparison between extrinsic and intrinsic measures, it seems that extrinsic measures add more proliferation resistance value than intrinsic features. More precisely, looking backwards, the combination of institutional measures and commitments under international treaties such as the NPT with verification efforts (i.e., safeguards) has already proven to be a highly effective extrinsic measure with significant impact on the global non-proliferation culture, whereas intrinsic features have played a minor role. This situation is fundamentally different for other factors such as physical protection, for example, where technical, i.e., intrinsic measures are clearly dominating.

Considering the importance of compliance verification and safeguards the question arises how this extrinsic measure can be strengthened in the future. In this context, both new approaches to international safeguards and new instrumentation that can support the treaty verification regime are of high interest. But not only future strengthening is critical; facilitating the implementation of new approaches is of equal importance. Keeping in mind that an appropriate mix of extrinsic and intrinsic measures is a preferred solution, the implementation of both should be jointly discussed. To focus the discussion in this area, the GIF Proliferation Resistance and Physical Protection Expert Group (PR&PP EG)[4] revisited the term “Safeguardability”, which will be discussed in more detail with a closer look on its impact on future safeguards instrumentation in the next section.

## Safeguardability

The GIF PR&PP EG captures the term *Safeguardability* as “the degree of ease with which a system can be effectively and efficiently put under international safeguards.”[5] Safeguardability is understood as a property of the whole nuclear system and applies to nuclear materials, process implementation, and facility design. The implementation of Safeguardability features is of special interest during the early development stage of nuclear facilities to facilitate the implementation of safeguards. More specifically, the Safeguardability attributes range from the convenience of performing Design Information Verification (DIV) and nuclear material accounting to the ease of implementing Containment and Surveillance (C/S) or other monitoring measures.

Selected examples for Safeguardability attributes number: the transparency of the facility layout, 3D scenario reconstruction models, comprehensiveness of facility data for DIV, uniqueness of materials signature or near real-time accountancy for nuclear material, and visual monitoring or automation for

C/S. Nuclear systems designers should be presented with a list of such attributes that can guide them in their efforts to take Safeguardability into account during development. The later implementation of safeguards can then be concluded in a more effective and efficient manner. Safeguardability, however, should not be understood as a string of necessary add-ons that are implemented in spite of operator concerns or design constraints. Rather, a well-evaluated approach that facilitates synergies with other system design features should be selected.

The critical question in the development of both Safeguardability attribute implementation and the safeguards instrumentation to support it is, "What will safeguards look like when such new nuclear systems are finally constructed?" Considering that the time between the design of nuclear installations and their eventual operation is measured in decades rather than years, the answer to this question is not necessarily straightforward. The last 15 years alone have seen significant changes in the safeguards and non-proliferation culture, with geopolitical events driving the implementation of the Additional Protocol (INFCIRC/540 corr.) and initiating discussions about multi-national fuel cycle models. If similar changes occur during the development of fourth generation nuclear reactors or multi-national approaches, the provision of Safeguardability attributes might be overtaken by events.

Challenges seem even more drastic when looking at the safeguards instrumentation that will support the future safeguards inspection regime as technologies change rapidly and new solutions become available at shorter time intervals. But safeguards instrumentation also offers a certain planning security. Since the inception of safeguards, for example, the physics for certain non-destructive assay applications have remained unchanged in principle, only varying in the instrumentation design. The same holds true for surveillance; the cameras and data acquisition algorithms have evolved, but the basic principle of using images to investigate events of interest has stayed the same.

Furthermore, safeguards instrumentation can facilitate synergies with other factors impacting nuclear fuel cycles and multi-national approaches such as personnel safety, quality assurance, and physical protection. Such synergy considerations, as well as the impact of a changing safeguards culture on instrumentation, will be investigated in the next section.

## Future Safeguards Instrumentation

The implementation of safeguards is not a static approach, but rather of a very dynamic nature with the flexibility to adapt to changes within the non-proliferation regime and treaty compliance verification efforts. One such transition that is currently on-going is driven by the implementation of the Additional Protocol and Integrated Safeguards. In the practical sense, this means that the safeguards system is shifting from a quantifiable declaration-and-verification regime to a more information-driven, qualitative approach. In an effort to verify both the correctness and the completeness of a NPT signatory state's declarations, traditional safeguards measures are re-evaluated and complemented by other information sources to detect undeclared materials and activities in addition to diversion and misuse of declared ones.

This has a direct impact on the instrumentation that safeguards inspectors deploy during their inspection visits. For traditional safeguards, instrumentation is designed for applications such as verification of declared material compositions, monitoring of specific operational activities (e.g., open core operations), and keeping materials and access points under seal. During Additional Protocol, or Complementary Access inspections, the nature of instrumentation that is required is fundamentally different. The inspector has limited or no knowledge about what to expect; therefore, the instrumentation required to support him/her must be portable and much more versatile than fixed installed monitoring systems or even the portable traditional systems that are designed to verify declared materials.

Further, the location where measurements or samples were taken during Complementary Access inspections is of critical importance for later analysis and cross-matching with other information sources such as satellite imagery, wide area monitoring, or open sources. This implies the need for better data management and location tagging capabilities, if possible. Fixed installed, unattended instrumentation will undergo changes as well, as new technological approaches become available and the shift towards an information driven, qualitative assessment allows for the drawing of state-level

conclusions about the absence of undeclared nuclear materials and activities in addition to the correctness of declarations.

The development of advanced and fourth generation reactor models has interesting consequences for safeguards instrumentation, as well. The implementation of safeguards measures during the design of such installations can alleviate the impact treaty verification efforts have on the operation of a nuclear installation today. To mitigate the need to pull cabling, retrofit the facility to provide the infrastructure for instrumentation, and optimize the inspection speed are all factors that will be appreciated by the operator.

In domestic safeguards systems, safeguards instrumentation does not necessarily have to operate for safeguards purposes only. There is a broad range of possible synergies, especially when the application of equipment is evaluated prior to the completion of the design of a facility. Surveillance cameras, for example, produce image data that IAEA inspectors use to draw conclusions about the correctness of declared and the absence of undeclared activities. Such image data are of interest to other concerns at a nuclear installation. First of all, it could be used to support physical protection measures as it might give an indication on insider or collusion threats. Next, it could strengthen personnel safety measures if image data analysis capabilities that can detect smoke or indication for other hazardous situations are added. Also, image data can provide a management tool if the operator can use image data to see if personnel are properly trained, rules are obeyed, and procedures (e.g., two-person rule) are followed.

In international safeguards there are concerns that the IAEA can not allow the operator to see exactly what data are used for safeguards verification purposes. However, new instrumentation could have the capability to generate different datasets specifically for each interested party that only contain the data necessary for their specific purpose. Such data would have to be independently authenticated to ensure their integrity. But if such requirements can be fulfilled, the same instrumentation could be utilized by multiple parties for various purposes.

With the shift of safeguards towards Integrated Safeguards and state-level conclusions, the question arises as to whether or not there will be a need for surveillance in future safeguards applications. Such discussions are mainly driven by the resources needed to operate a surveillance infrastructure not only for the equipment, but also for the image data analysis, field maintenance and support, and the frequency with which their data need to be extracted and reviewed. If multiple parties shared the benefits of surveillance, however, it could advance to be a feature implemented easily during the design with its cost shared among the users, thus becoming a true Safeguardability benefit.

Also, other instrumentation can be envisioned for synergies with new safeguards approaches. New measurement techniques that replace swipe sample taking and allow for immediate rather than destructive analysis could be added to the safeguards portfolio. Following the shift towards information-driven safeguards, such technologies can be envisioned in a portable form, as well. As an example, laser spectroscopy measurement techniques can be deployed to immediately detect and analyze the presence and enrichment of UF6 in a given air sample. Such a technology could be used in portable applications to detect undeclared enrichment programs at undeclared sites or enrichment higher than declared at declared facilities. But it could also be employed in a fixed installation for continuous, on-line measurement.

If the measurement accuracy of such an approach is comparable or better than the currently used mass spectrometry, safeguards authorities will not be the only parties interested in it. Facility operators will have a similar if not larger interest in using the same technology for their quality assurance and cost-effectiveness qualities. Again, synergies between multiple users can be realized, the implementation facilitated during the design of the instrumentation, and the cost shared among the beneficiaries; thus truly offering Safeguardability attributes.

## Conclusions

Safeguardability and the shift towards information-driven safeguards is a complicated concept that bears both challenges and opportunities. The implementation of Safeguardability attributes can not be unconditionally requested from nuclear installation designers. Their benefits need to be carefully

balanced against implementation difficulty and cost. Only if a benefit exists for both sides, treaty compliance verification authorities and operators, the implementation will be possible. Similar considerations apply for the instrumentation that will support Safeguardability and future safeguards. Only if joint use, data sharing, and synergies can be realized while all security and data integrity concerns are addressed, the instrumentation will be a valuable addition for all parties involved.

Decisions on how to best proceed towards the new safeguards regime can not be made by safeguards authorities alone. Rather, the early involvement of all participants to jointly decide on a course of action will promise the greatest chances of a rewarding result. This also needs to be a continuous process. As quantitative elements decline and qualitative elements increase, careful discussion of all stakeholders is needed to adapt existing agreements to changes in the non-proliferation regime and to the availability of new technologies. Also, what might be identified as an approach with high synergies between operators and safeguards for new nuclear reactors might not be applicable for existing facilities if cost and effort of retrofitting exceed the advantages of new instrumentation.

In support of new, proliferation-resistant fuel cycles and multi-national approaches, the goal should be to set a new standard for future nuclear safeguards while carefully measuring the interdependencies with other critical factors such as physical protection, environmental concerns, personnel safety, quality assurance, and economic sustainability. Only a balanced approach with input from all stakeholders can facilitate a swift and synergetic implementation.

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[3] 'Technological Opportunities to Increase the Proliferation Resistance of Global Civilian Nuclear Power Systems,' TOPS Task Force of the Nuclear Energy Research Advisory Committee, http://www.ne.doe.gov/pdfFiles/TOPS-Final.pdf, 2001.
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[5] Ibid.

# Non-proliferation aspects of advanced fuels under light water reactor conditions

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***Abstract:***

*In the paper, different nuclear fuel cycles of advanced fuel types and their non-proliferation aspects are examined and compared. The investigated fuels include mixed oxide fuel, thorium based fuel and zirconium inert matrix fuel. All of them are used to carry and burn or transmute plutonium created in the classical UOX cycle. The computing cycles are based on reprocessing of spent UOX fuel, separation of plutonium, fabrication of an advanced fuel type and its reuse in a light water reactor. Minor actinides are separated along with plutonium only in the case of the inert matrix fuel. The calculated and compared values include plutonium and minor actinides transmutation rates, mass of reprocessed fuel and mass of fuel sent to the repository. All fuel cycles were calculated by HELIOS 1.9 spectral code.*

**Keywords:** plutonium transmutation, MOX fuel, inert matrix, thorium based fuel

## 1. Introduction

Installation of advanced nuclear fuel cycles into operating conditions needs consistent studies of fuel material composition changes under neutron irradiation. This work examines plutonium and minor actinides changes using advanced nuclear fuel in the light water reactors.

The analysed cycles are based on reprocessing of the spent UOX fuel burned in the VVER-440 reactor type under normal operating conditions, separation of plutonium, fabrication of an advanced fuel type and its reuse in the same reactor type, VVER-440. Minor actinides are separated along with plutonium only in the case of the inert matrix fuel. Detailed information about the cycles is in the sequel.

The calculated and compared values include plutonium and minor actinides transmutation, the mass of reprocessed fuel and mass of fuel sent to the repository. All fuel cycles were calculated by HELIOS 1.9 spectral code.

## 2. Advanced fuel types and their cycles

Advanced fuel types bring several advantages in comparison with classical uranium fuel $UO_2$ used nowadays worldwide. The advance of novel fuel types is in their ability to transmute plutonium and minor actinides to non-active nuclei or to nuclei with a shorter decay time and in their non-proliferation resistance [1]. This work is focused on transmutation potential of mixed oxide fuel, inert matrix and thorium based fuel. All of them should be operated in power reactors in similar conditions to UOX fuel.

## 2.1. Mixed oxide fuel cycle

Mixed oxide fuel (MOX) is a well known type of fuel prepared by mixing separated plutonium oxide with uranium U-238 with a small content of uranium U-235. The analysed MOX fuel cycle is as follows: Natural uranium is enriched and burnt in a light water reactor in the same way as in the case of the open fuel cycle (OFC) to target burn-up 50000 MWd/tHM. After a cooling time of 5 years the spent fuel is reprocessed and plutonium isotopes are separated. The reprocessing calculates with 0.1% plutonium loses. Separated plutonium is then mixed with depleted uranium U-238 (0.25% of U-235). To reach similar multiplication ability as with UOX fuel, the content of plutonium is set to 8.5%. The plutonium isotope vector in spent UOX fuel is summarized in Table 1, MOX fuel cycle is in Figure 1.

| *Pu isotope* | *% composition* |
|---|---|
| Pu-238 | 2.78 |
| Pu-239 | 55.46 |
| Pu-240 | 23.20 |
| Pu-241 | 12.16 |
| Pu-242 | 6.40 |

**Table 1:** Plutonium isotopes content in the plutonium vector of the spent UOX fuel.

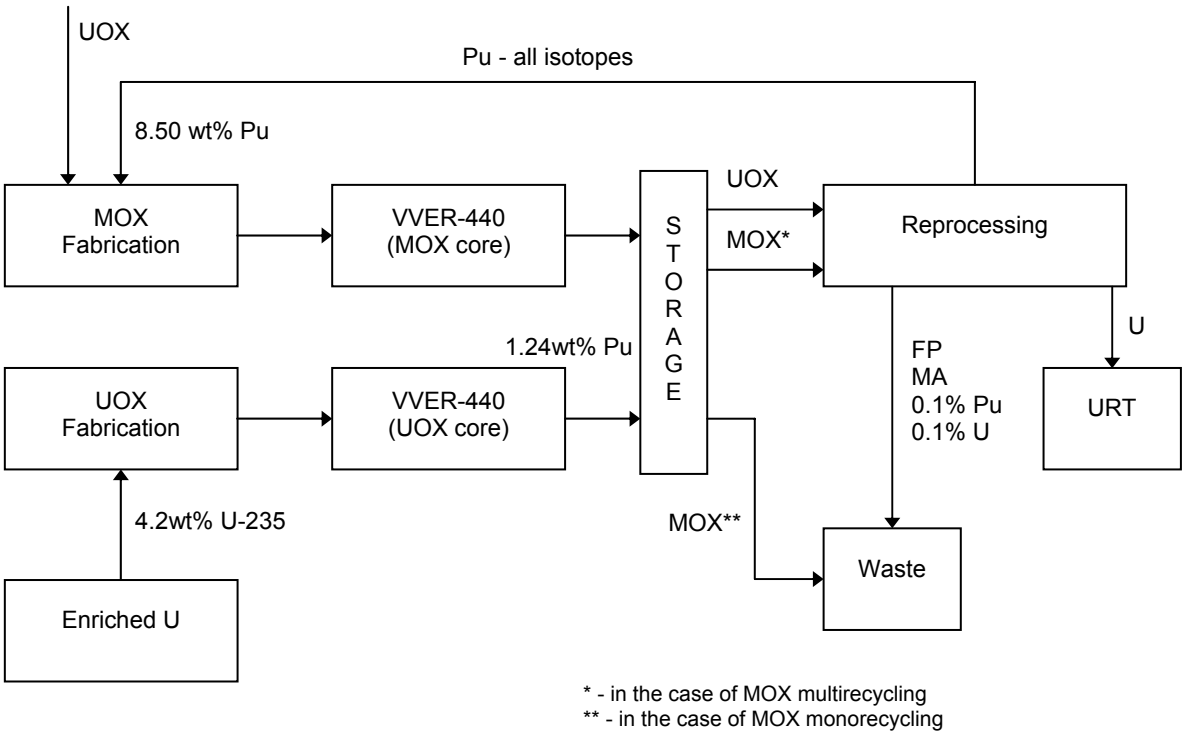


**Figure 1:** MOX fuel cycle.

The fuel cycle is analysed in two different variants. The first one is monorecycling of plutonium, which means that plutonium is separated only once from UOX spent fuel and burnt in MOX fuel. Burnt MOX fuel is then considered as waste and stored. The second case is multirecycling of plutonium to the equilibrium state. Burnt MOX fuel is reprocessed and plutonium isotopes are separated and burnt again in fresh MOX fuel until there is no material difference between burnt MOX fuel from two subsequent cycles.

Multirecycling of plutonium in MOX fuel causes changes of the plutonium vector during the cycles. The vector of plutonium in MOX equilibrium state is in Table 2. The content of plutonium in MOX fuel for monorecycling and also for multirecycling of plutonium was set as mentioned above to 8.5%, plutonium isotopes vectors were set according to Table 1 for monorecycling of plutonium, and according to Table 2 for multirecycling of plutonium.

| *Pu isotope* | *% composition* |
|---|---|
| Pu-238 | 0.68 |
| Pu-239 | 11.41 |
| Pu-240 | 7.12 |
| Pu-241 | 1.13 |
| Pu-242 | 79.65 |

**Table 2:** Plutonium vector in MOX equlibrium cycle.

## 2.2. Thorium based fuel

The second analysed advanced fuel type is thorium based fuel with plutonium content (ThPu). The fuel cycle is similar to MOX fuel, the difference is only in mixing the separated plutonium oxide with thorium oxide. The case of monorecycling is analysed. Plutonium content in thorium fuel was estimated to reach values of the multiplication factor similar to that reached in the case of UOX open fuel cycle. Several plutonium contents were calculated and finally 5.40% of plutonium Pu-239 content was chosen for the cycle. The other isotopes of plutonium which are present in the spent UOX fuel are in proportion to Table 1. The total amount of plutonium in the fuel is 9.74%. The scheme of the fuel cycle is in Figure 2.

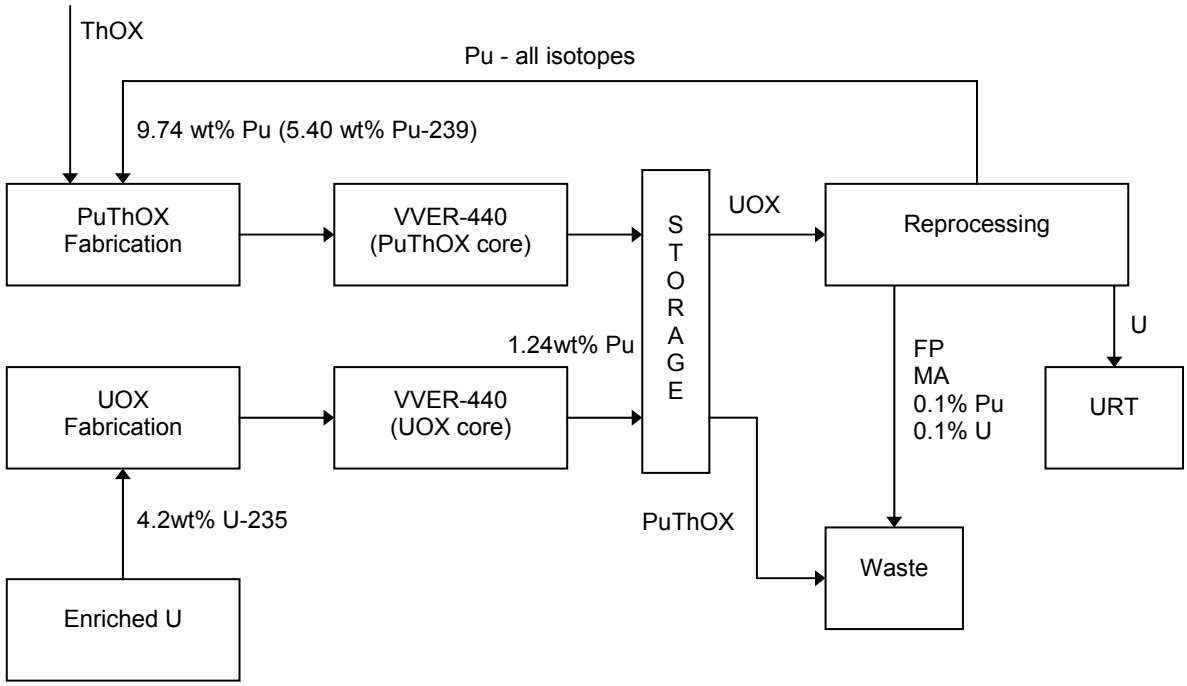


**Figure 2:** Thorium based fuel cycle.

## 2.3. Inert matrix fuel cycle

Inert matrix fuel (IMF) is a fuel prepared by mixing separated plutonium and minor actinides into an yttria stabilized zirconium matrix. The advance of the inert matrix fuel is in the non-proliferation resistance against outside impacts. The cycle is similar to MOX fuel cycle: Natural uranium is enriched and burnt in a light water reactor in the same way as before to target burn-up 50000 MWd/tHM and after a cooling time of 5 years the spent fuel is reprocessed. Separated plutonium and minor actinides are then mixed with the zirconium inert matrix and loaded into a fresh assembly. To ensure symmetrical distribution of power loading, a new type of fuel assembly was modelled. Detailed information about computation models is given in the next chapter.

The analysed fuel cycle with inert matrix fuel is operated in the self-cleaning manner. Separated plutonium and minor actinides from one burnt UOX assembly are loaded into one advanced assembly to selected pins. The cycle calculates only with 0.1% of Pu and MA losses during the separation process. There is no multirecycling of the inert matrix fuel. A scheme of the advanced cycle is in Figure 3.

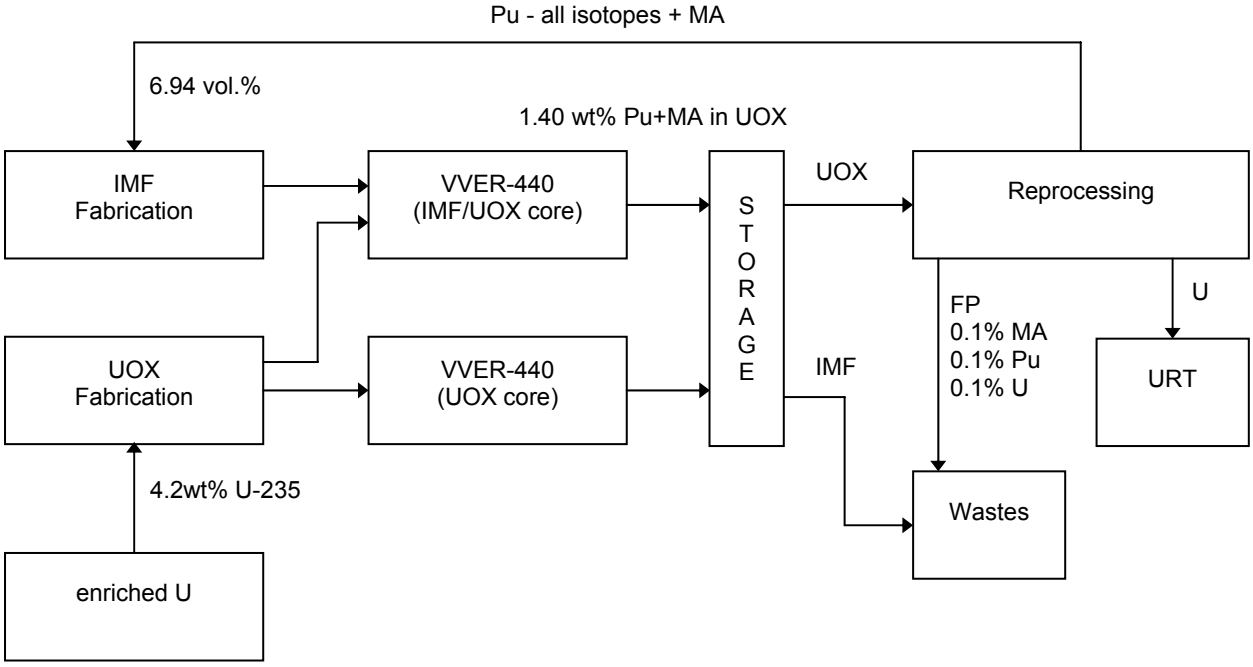


**Figure 3:** Inert matrix fuel cycle.

More information about cycles can be found in ref. [2], [3].

## 3. Computation models

Two models of VVER-440 assemblies were prepared. The first one, Figure 4a, is the VVER-440 assembly with one type of fuel pins. This assembly was used for calculations of MOX fuel cycle and thorium based fuel cycle. To perform calculations with inert matrix fuel an advanced VVER-440 assembly was prepared, Figure 4b, with two different fuel types. The advanced fuel is placed into dark pins, the rest are fresh UOX pins.

The assemblies are computed in an infinite lattice – neutrons which escape from the one surface of the assembly and enter the assembly at the other one. The models were prepared and fuel cycles were calculated by HELIOS 1.9 spectral code [4].

Target burn-up is the same for all types of fuels, 50000 MWd/tHM in 5 cycles of 320 days.

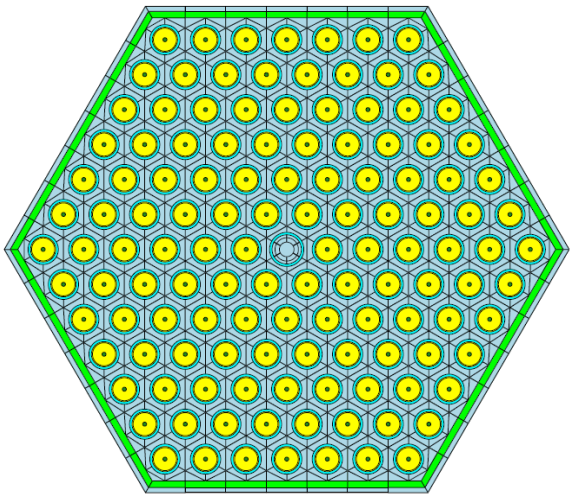

**Figure 4a:** Model of VVER-440 assembly

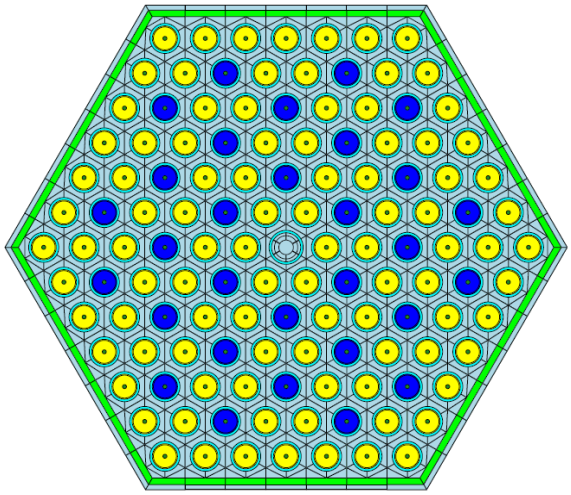

**Figure 4b**: Model of advanced VVER-440 assembly

## 4. Results

Equilibrium advanced fuel cycle calculations were performed. Comparison of the reached multiplication factors in different fuel cycles with the open fuel cycle is in Figure 5.

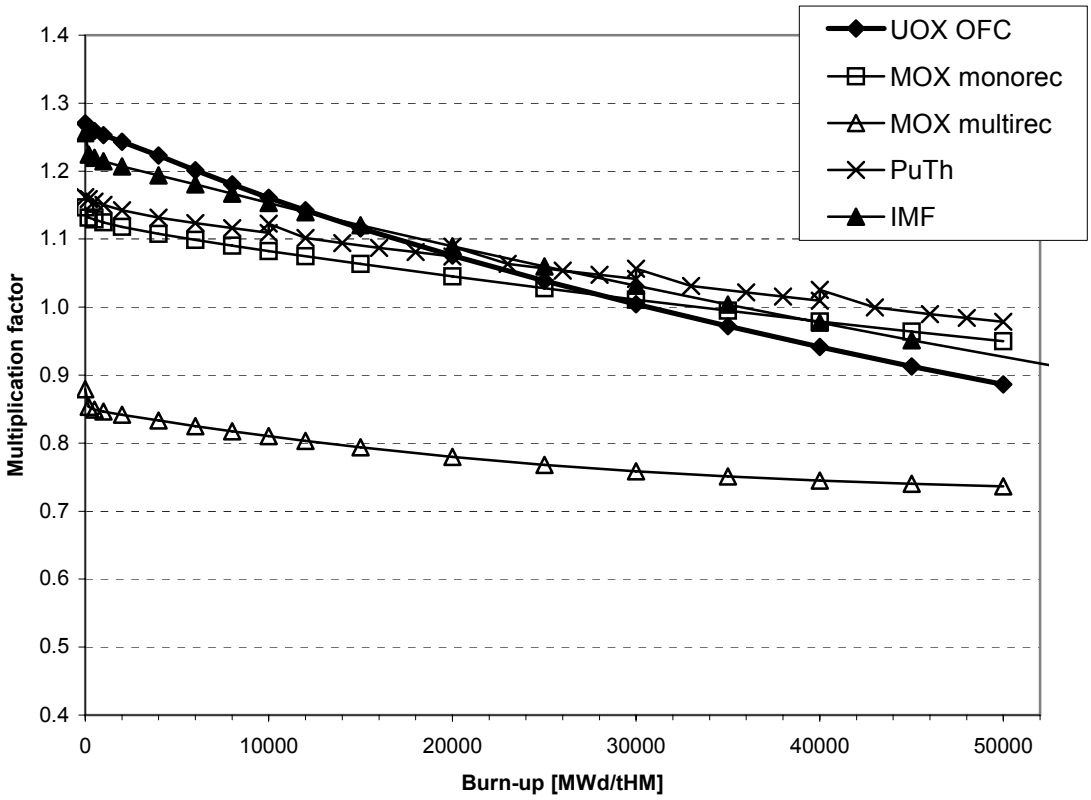


**Figure 5:** Multiplication factor for different fuel cycles in comparison with UOX open fuel cycle

From Figure 5 it can be seen that multirecycling of MOX fuel cannot be operated in a light water reactor. The values of the multiplication factor for this equilibrium cycle are very low. It is

recommended to perform three recycling steps maximally and then consider the burned MOX fuel as waste. To prepare advanced nuclear fuel for one reactor more than one UOX reactor are needed due to a higher plutonium content as is in spent UOX fuel. It is only the inert matrix fuel cycle that works in the self-cleaning cycle, which means that all plutonium created in one UOX/IMF core is separated and recycled to fresh IMF fuel for one UOX/IMF core. The numbers of feeding reactors for other cycles are not negligible.

Table 3 summarizes the total number of UOX reactors needed to operate the equilibrium cycle. In the case of burning weapon plutonium or plutonium stored in spent fuel worldwide, no feeding reactor is needed to operate these fuel cycles. Table 3 also summarizes the initial content and content of plutonium and minor actinides in the spent fuel and transmutation rates for Pu and MA, masses of fuel entering reactors and masses of fuel sent to repository with the total amount of finally disposed plutonium.

| | *UOX* | *MOX monorec* | *MOX multirec* | *PuTh* | *IMF* |
|---|---|---|---|---|---|
| *Number of feeding reactors* | - | 6.82 | 6.82 | 7.87 | - |
| *Pu and MA initial content [wt %]* | 0.00 | 8.50 | 8.50 | 9.74 | 1.41 |
| *Pu and MA content in burned fuel [wt %]* | 1.22 | 6.74 | 8.33 | 5.75 | 0.46 |
| *Pu transmutation rate [kg/TWhe]* | 0.00 | 51.06 | 35.26 | 104.44 | 30.72 |
| *MA transmutation rate [kg/TWhe]* | 0.00 | 0.00 | 0.00 | 0.00 | 0.47 |
| *Mass of reprocessed fuel [t/TWhe]* | 0.00 | 2.40 | 2.40 | 0.00 | 2.39* |
| *Average quantity of separated Pu [kg/TWhe]* | 0.00 | 118.63 | 141.38 | 221.17 | 41.19 |
| *Amount of finally disposed Pu [kg/TWhe]* | 26.02 | 118.63 | 141.38 | 116.73 | 10.48 |

* - UOX from UOX/IMF core is reprocessed

**Table 3:** Comparison of selected parameters of advanced nuclear fuel cycles.

The total amount of finally disposed fuel is a little bit misleading. To operate one MOX reactor, almost 7 UOX reactors are needed (to produce enough plutonium to reach 8.5% plutonium content in the fresh MOX fuel) from which all plutonium, except 0.1% losses, is separated and used in MOX fuel. Hence, the higher values of the total amount of disposed plutonium include plutonium from these feeding reactors. In the case of burning plutonium from spent fuel storage or weapon plutonium these values should be lower in comparison with UOX open fuel cycle.

## 5. Conclusions

An overview of several advanced nuclear fuel cycles taken into account in sustainability evaluations is given. The total amount of disposed plutonium and high level waste can be reduced by introducing of advanced fuel cycle into power reactors.

From the point of view of the total amount of finally disposed plutonium, the inert matrix fuel cycle seems to be the best choice for the equilibrium cycle.

Advanced fuel types bring higher proliferations resistance against impacts from outside. The problem of proliferation is in the step of separation of plutonium and adding of plutonium to the fuel matrix. The highest volume of plutonium is separated in the thorium fuel cycle, the lowest one in the case of the inert matrix fuel. From this point of view the inert matrix fuel cycle is also the best choice [5].

Advanced fuel types can be operated under light water reactor conditions. They can participate in transmutation of cumulated plutonium and also in plutonium production reduction. Introduction of advanced cycles into power reactor practice requires detailed studies of operation and other characteristics.

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# Proliferation Resistance and Physical Protection Robustness Characteristics of Innovative and Advanced Nuclear Energy Systems

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**Abstract**

*Since the early seventies, a lot of effort has been put in trying to define/evaluate the proliferation resistance of nuclear energy systems and their associated nuclear fuel cycles. Past studies put in evidence how it wasn't possible to conceive a proliferation-free nuclear fuel cycle (hence the need of a suitable safeguard system), but also stressed that not all the available options are equivalent.*

*The topic has become of renewed interest in the context of the innovative reactor and nuclear energy systems design concepts, presently under development. New reactors will have to exhibit and demonstrate enhanced features with respect to the existing ones.*

*It is common practice to classify Proliferation Resistance (PR) and Physical Protection (PP) characteristics of a system in intrinsic, i.e. belonging to the system, and extrinsic, such as those related to the application of international safeguards.*

*This paper will summarise in a critical way some of the Proliferation Resistance & Physical Protection (PR&PP) intrinsic features as emerged so far in a number of studies and reports available in this field and can contribute to provide a first input to designers to brainstorm on a number of possible requirements. This survey is part of JRC activity in contribution to Generation IV International Forum (GIF).*

**Keywords:** Proliferation Resistance, Physical Protection Robustness, Intrinsic characteristics.

## 1. Introduction

Proliferation resistance of innovative Nuclear Energy Systems (NES) and their associated fuel cycles is being reconsidered after various studies developed in the 70s [1].

The Generation IV International Forum (GIF) aims at developing a set of promising reactor concepts to be studied and developed to be ready for deployment in the years 2020-2030. Proliferation Resistance & Physical Protection, Safety, Economics and Sustainability are indeed the four goal areas where innovative nuclear energy systems will have to excel, according to the Generation IV International Forum roadmap project report developed in 2002 [2].

Both for PR and PP, it is common practice to distinguish between intrinsic characteristics of the system), due to its design, lay-out and interfaces, and extrinsic measures, related to local normative and to the addition of international safeguards to the system [3].

The present paper will summarise in a critical way some of the PR and PP intrinsic features of innovative nuclear energy systems, emerging from a number of studies and reports available in this field (GEN-IV Forum, IAEA, other scientific publications).

This survey is done in the context of JRC-IPSC-NUSAF's contribution to GEN-IV PR&PP Expert Group.

## 2. PR and PP Intrinsic Characteristics

According to [3] and to the definitions taken from GEN-IV PR&PP Methodology Rev.5 [4]:

- ***Physical protection (robustness)*** *is that characteristic of an NES that impedes the theft of materials suitable for nuclear explosives or radiation dispersal devices (RDDs) and the sabotage of facilities and transportation by sub-national entities or other non-Host State adversaries*

- ***Proliferation resistance*** *is that characteristic of an NES that impedes the diversion or undeclared production of nuclear material and the misuse of technology by the Host State seeking to acquire nuclear weapons or other nuclear explosive devices.*

Physical Protection requires adequate security of the energy production plants and of related fuel cycle plants, but also by the adoption of designs that rely as much as possible on passive safety systems, able to operate without electric power as it might occur in case of sabotage.

Proliferation resistance is a requirement, that must be insured for the whole fuel cycle and not only for the reactor, and can be technically satisfied through several solutions, e.g. irradiated fuel burn-up; characteristics of plutonium in the spent fuel and its possible separation in a reprocessing plant, if any; full core management by a reliable vendor.

Intrinsic PR features depend on the strategic choices for the system to develop (e.g. reactor type, fuel cycle, material qualities) and the design adopted to cope with technical requirements and difficulties. They play therefore a key role in proliferation resistance, both in making the system a non-attractive route to diversion, and in facilitating the implementation of safeguards. See to this respect the Appendix D, in the Addendum of the PR&PP methodology study report [5], tackling the safeguardability issue of advanced nuclear energy systems and the paper in the proceedings of this Symposium focussing also on Safeguardability [6].

The main intrinsic PR features of a nuclear system to be considered have been identified by GIF [2], international activities like IAEA/INPRO [7], as well as by other studies [8].

These include design features that can increase technological difficulties for diversion of fissile material and fabrication of weapons, like:

- Type, accessibility and inventory of feed fuel
- Evidence of separated fissile material throughout the fuel cycle, which is linked to the reprocessing process
- Spent fuel characteristics (e.g. burn-up, radiation barriers, isotopic composition, heat generation rate, neutron emission, critical mass, radiation signature for detectability)

Fissile material could be subtracted by the system at any stage, i.e. as fresh or spent fuel element in reactor, during re-processing, if applied, or even during transport to re-processing or final conditioning before disposal in the open-cycle case.

## 3. PR&PP Features from GIF Roadmap

As mentioned above, Generation IV reactors will have to excel with respect to the existing second and third generation ones. Different reactor system concepts and the associated nuclear fuel cycle options were considered during the GEN-IV *Roadmap project* (2001-2002) [2].

The following four goal areas of excellence were defined for Generation IV nuclear energy systems:

1. Sustainability, (SU);
2. Economics, (EC);
3. Safety and Reliability, (SR);
4. Proliferation Resistance and Physical Protection (PR)

The four goal areas of excellence were assigned to eight equally important goals:

- Resource Utilisation (SU1)
- Waste Minimization and Management (SU2)
- Life Cycle Cost (EC1)
- Risk to Capital (EC2)
- Operational Safety and Reliability (SR1)
- Core Damage (SR2)
- Offsite Emergency Response (SR3)
- Proliferation Resistance and Physical Protection (PR1)

These were worked out into 15 suitable weighted criteria and, finally, into 26 metrics. Four GIF Technical Working Groups (TWG), one per each reactor system type, evaluated a first round of 124 Innovative Nuclear Systems.

The evaluation was done based on a comparison between the metric of the system's features and a reference value typical of 3rd generation systems (Advanced Light Water Reactor, ALWR).

After a first screening process, 19 concepts were selected to be further scrutinized in detail and evaluated according to an evaluation process developed in course of the Roadmap study [9]

For what concerns the evaluation of the Proliferation Resistance and Physical Protection (PR&PP) of the Generation IV proposed nuclear energy systems, the hierarchy of goals, criteria and metrics is that reported in table 1.

| Goal Area | Goal | Criteria | Metrics |
|---|---|---|---|
| Proliferation resistance and physical protection | PR1 Proliferation resistance and physical protection | PR1-1 Susceptibility to diversion or undeclared production. | Separated materials |
| | | | Spent fuel characterization |
| | | PR1-2 Vulnerability of Installations. | Passive safety Features |

**Table 1**: PR&PP Roadmap Evaluation criteria and metrics.

Although openly available on the worldwide web, most of this material was disseminated only through the web and not reported at conferences.

A huge amount of material has been generated during the application of the evaluation process, at the end of which the six nuclear energy systems to be studied in the forthcoming years were identified. The next paragraphs will describe the various types of innovative reactors and their relevant PR features, focussing on the six GEN-IV reference designs.

## 4. Summary of Innovative Reactor Designs

A large number of innovative reactor systems is being studied worldwide. Different designs have the chance to be actually developed, as the possible future reactor fleet will be formed by different complementary types of nuclear systems, able to exploit as much as possible the fissile and fertile properties of uranium, and possibly thorium, at the same time minimizing waste generation and its radiological issues by recycling minor actinides. With all non proliferation issues to be taken into account.

The classification of innovative nuclear energy systems is done according to the reactor types:

- Water cooled

- Gas cooled
- Liquid Metal cooled
- Non-conventional

## 4.1. GIF Roadmap's NES

As reported in [10, 11, 12, 13] the four GIF Technical Working Groups of experts analysed and screened a total of 38 innovative Water Cooled, 21 Gas Cooled and 33 Liquid Metal cooled and 32 non-conventional innovative nuclear energy systems of various sizes.

The first screening against the fifteen weighted criteria lead to the selection of the following 19 nuclear energy system concepts, listed with their main differences, from which were eventually identified six reference systems, with various options, marked in italic.

### 4.1.1. Water Cooled reactor systems

W1-LWR Integral Primary System Reactor Concept Set
W2-Large Simplified Boiling Water Reactor (ESBWR)
W3-NG (Next Generation) CANDU - With Low Enriched Uranium (LEU) Once-Through Cycle
*W4-SCWR Supercritical Water Reactor (SCWR) – Thermal Spectrum*
*W5-SCWR Supercritical Water Reactors - Fast Spectrum (SCWR-Fast)*
W6 High Conversion Advanced Boiling Water Reactor, ABWR-II

### 4.1.2. Gas-Cooled reactor systems

G1 PBR Modular Pebble Bed Reactor - Once Through
G2 PMR Prismatic Fuel Modular Reactor - LEU Open Cycle
*G3 VHTR Very High Temperature Reactor - LEU Open Cycle*
G4-Generic HTGR - Closed Synergistic Flexible Fuel Cycle
*G5- Gas-Cooled Fast Reactor GFR - Closed Cycle*

### 4.1.3. Liquid Metal Cooled (Fast Spectrum) reactor systems

*L1-SFR Sodium Fast Reactor, MOX Fuel, 1500 MWe*
*L2-SFR Sodium Fast Reactor, Metal Fuel, 760 MWe*
*L4-LFR Medium Pb/Pb-Bi Cooled, US Design Systems, 300-400 MWe*
*L5-LFR Large Medium Pb/Pb-Bi Cooled, Russian Design Systems, 1200 MWe*
*L6-LFR Small Pb/Pb-Bi Cooled, 50-150 MWe*

### 4.1.4. Non-Conventional reactor systems

*N1-MSR Molten Salt Liquid Core Reactor, Thermal & Epi-thermal spectrum, 1000 MWe*
N2-Liquid Core Reactor Systems
N3-Molten Salt Cooled Reactor Systems (AHTR), 1000 MWe

## 4.2. IAEA Study on Small and Medium Sized Reactors

Besides the designs considered by GIF's Roadmap, about 50 Small and Medium Reactor (SMR) concepts remain under consideration in more than 15 IAEA Members States. Small reactors have equivalent electric power less than 300 MWe, whereas medium ones are in the range 300-700 MWe.

A recent Technical Document issued by the IAEA [14] dedicated to the status of Small and Medium sized Reactor designs (SMRs) in 2005, describes a total of 13 Water Cooled, 6 Gas Cooled, 6 Liquid Metal cooled and 1 Non-Conventional reactor designs. Half of the design concepts presented in the document also appears in GIF's lists. Hereinafter follow descriptions of the main technical and PR&PP characteristics of the four groups of reactor designs, as taken from GIF's Roadmap [2, 10-13] and IAEA's lists [14].

# 5. Innovation in fuel re-processing

Before entering in details of the innovative designs, it is worthwhile to briefly recall the two main types of fuel re-processing, with different PR implications, which will be mentioned throughout the next paragraphs in relation to the fuel type.

## 5.1. Advanced Aqueous

Evolving from traditional PUREX, the Advanced Aqueous process is characterized by uranium and plutonium co-extraction, along with most of the minor actinides, and no separation of plutonium at any stage of the process. This increases proliferation resistance of the system because the processed material is low-decontaminated by the presence of Minor Actinides.

This results in too high a radiation activity to require fuel fabrication in a simple glove-box facility, which complicates the operations, but conversely enhances proliferation resistance by a higher radiological barrier. Moreover, due to the shorter half-life of fission products compared to the heavier actinides that are recycled and burned, waste radiotoxicity is reduced in time and inventory up to a factor of 100 [15],[16].

In comparison to this improved method, by traditional PUREX U and Pu are separated with an industrial yield close to 99.9%, while Minor Actinides (MA) and Fission Products (FP) are conditioned in a glass matrix for interim storage and final disposal.

## 5.2. Pyro-metallurgy

Developed since the 80's, this process has only reached the pilot-scale stage, and it is claimed to be more compact, less complex, less costly and generating less waste streams than conventional aqueous (PUREX) process used for oxide fuel [12].

Pyro-processing of metal fuel is based on few process steps by electro-chemical dissolution (electro-refining) in a molten salt eutectic, U, Pu and other actinides are co-extracted, and there is no recovery of pure fissile material at any stage of the process.

Moreover, recycled fuel needs to be remotely fabricated because of the inherently low decontamination factors, which complicates operations and enhances proliferation resistance.
Pyro-metallurgical methods can be applied also to MOX fuels [17].

# 6. Water-Cooled Reactor Systems

38 Water-cooled reactor systems and fuel cycle concepts were considered by GIF's Roadmap TWG1 [10]. These also included most concepts belonging to the largest family of SMRs (thirteen concepts, 50% of the total), the majority of which is formed by Light-Water type reactors (six Pressurized Water Reactors, three boiling water reactors (BWR), one indirect BWR, two innovative pool type ones) and one Advanced Heavy Water Reactor [10].

Some water-cooled SMRs like SMART[1], IRIS, MARS, IMR present longer operational cycles and reduced number of inspections, which could be seen as simplifying the implementation of safeguards [14]. IRIS is characterized by regional or centralized reprocessing, a burn-up which at a later stage could attain 120 GWd/tHM and degradation of secondary plutonium isotopic composition. Beside the interesting VBER-300 floating NPP, also noticeable are designs foreseeing a closed nuclear fuel cycle like the RMWR. AHWR's (Advanced Heavy Water Reactor) basic design is characterized by a once-through with Pu /Th /U-233 fuel cycle, but can also be extended to a closed fuel cycle [14].

## 6.1. GEN-IV Supercritical-Water-Cooled Reactor System (SCWR)

Amongst Water Cooled reactors, GIF's selection prioritized supercritical fission reactors, to be developed by 2025, whose characteristics are the following:

[1] For the meaning of the systems acronyms, see the acronyms list at the end or the paper.

- Operation at high temperatures and pressures (above the light water critical point of 374 oC, 221 bar)
- High thermal efficiencies (up to 45%)
- Very compact nature of the physical plant, with lower coolant mass inventory
- Simpler design than LWRs (no steam separators and generators)
- Higher heat transfer rate per unit mass flow (large specific heat above the critical point)
- Single-phase fluid with no re-circulation
- Both direct and combined direct/indirect cycles
- Both light and heavy water moderated concepts
- High coolant outlet temperatures allowing potential for hydrogen production
- Thermal or fast neutron spectrum

Supercritical water reactors can also be designed to operate as fast reactors. The difference between a thermal and a fast supercritical water-cooled reactor is in the lattice pitch and in the use of moderating material. The fast spectrum reactors use a tight lattice and no additional moderator material, whereas the thermal spectrum reactors need both a loose lattice and additional moderator material in the core. The thermal option foresees an open once-through cycle, while advanced aqueous reprocessing at a central fuel cycle facility is foreseen in the case of the fast option with MOX fuel.

### 6.2. PR Intrinsic features

Water-cooled innovative reactors present features similar to existing PWR and BWR:

- Low enrichment uranium dioxide fuel in fresh and spent fuel
- Once-through fuel cycle (most designs)
- Unattractive isotopic composition of plutonium in discharged fuel
- Radiation barriers provided by the spent fuel

The option of a closed U-Pu cycle applies to some of them, namely the fast ones, with issues related to advanced aqueous re-processing (see par. 5.1).

### 6.3. PP Intrinsic features

Safety features are in general similar to the reference ALWR system. Passive safety of SCWR is facilitated by the lower heat content of the reactor coolant system, which results in lower containment loadings during a design-basis Loss-Of-Coolant Accident (LOCA).

## 7. Gas Cooled Reactor Systems

Twenty-one high-temperature GCR system concepts were contributed to GEN-IV Roadmap's TWG-2 [11], grouped into:

- Modular Pebble Bed Reactor Systems (PBRs)
- Prismatic Fuel Modular Reactor Systems (PMRs)
- Very-High-Temperature Reactor Systems (VHTRs)
- Gas-Cooled Fast Reactor Systems (GFRs)

Key design characteristics of both pebble (PBR) and prismatic (PMR) gas-cooled systems are the use of inert helium coolant, graphite moderator with high strength and stability to high temperatures and fuel made of refractory ceramic-coated TRISO micro-particles. PBR, PMR and GFR should be based on a direct Brayton gas turbine cycle.

### 7.1. GCR Fuel

TRISO-coated particles (650 microns to about 850 microns) consist of a spherical kernel of fissile, or fertile material, encapsulated in multiple coating layers. The multiple coating layers form a highly

corrosion-resistant barrier, essentially impermeable to the release of gaseous and solid fission products up to high temperatures of 1600 $^{o}$C.

Two options exist for the fuel design:

- Prismatic block fuel, where TRISO-coated particles are mixed with a matrix and formed into cylindrical fuel compacts
- Pebble fuel, where TRISO-coated micro-spheres are contained in a 6 cm ball configuration called pebble.

Some pebble bed gas cooled reactor concepts rely on continuous re-fuelling within an annular core or in channels (PBR, APBR types), whereas others employ prismatic pin-in-block fuel (GT-MHR types) in graphite channels. With this fuel arrangement, GCRs can accommodate a wide variety of mixtures of fissile and fertile materials without any significant modification of the core design. The solid moderator in GCRs also avoids the positive void coefficient of reactivity, which limits the plutonium content of LWR MOX fuels.

The operating characteristics of the GCRs accommodate use of a wide range of fuel cycles without changing the basic reactor system design. The applicable fuel cycles range from LEU to thorium-uranium to plutonium alone. An option of closed fuel cycle could be foreseen for the GT-MHR, with MOX or hybrid U-Th fuel.

## 7.2. GEN-IV Very-High-Temperature Reactor System (VHTR)

Based on PBMR or GT-MHR concepts, but with a higher coolant's outlet temperature above 950°C, it is an advanced, high-efficiency reactor system, which can be used in energy-intensive, non-electric processes, hydrogen production as well as supply process heat to a broad spectrum of high temperature applications. Its main features are:

- High thermal efficiency
- High burn-up, hence reduced waste production and disposal burden
- Larger scope of potential waste applications, for example, coal gasification and metallurgic processes
- Improved intrinsic proliferation resistance due to refractory coated fuel, low fissile inventories and open fuel cycle
- Increased passive safety due to refractory fuel precluding damage under all operating and accident conditions
- Thermal neutron-spectrum and once-through uranium cycle
- Flexibility to adopt U/Pu fuel cycles and improve waste minimization

VHTR is the nearest-term hydrogen production system, foreseen by 2020, but still needs R&D on high-temperature resistant alloys, fibre-reinforced ceramics or composite materials, and zirconium-carbide fuel coatings, including confirmation of fuel behavior under accident conditions.

## 7.3. GEN-IV Gas-Cooled Fast Reactor System (GFR)

Gas-cooled-Fast reactors with a direct Brayton cycle are potentially high sustainable and economically competitive, combining high temperature with breeding fuel and burning actinides. They exhibit the same safety features of thermal gas-cooled reactors, with the advantages of a closed and integrated fuel cycle minimizing the needs for mining, transports of nuclear materials, and proliferation issues.

The reference concept is a 600 MWth/288 MWe, helium cooled reactor system operating with an outlet temperature of about 850°C, using a direct Brayton cycle gas turbine. The thermal efficiency is estimated around 48%, with potential for hydrogen production.

GFR has three fuel design options, including both the prismatic with fuel particles or composite fuels, and fuel pins with actinide compound/solid solution. Besides ensuring high fission gas retention up to 1600 $^{o}$C, R&D should support the achievement of high burn-up (up to 15% FIMA).

The foreseen GFR development is by 2025, with a cycle lifetime of 5-10 years, and a global breeding ratio greater than one.

## 7.4. PR Intrinsic features

Intrinsic proliferation resistance features common to this category include:

- Once-through fuel cycle (except GFR)
- High fuel burn-up, with low residual plutonium inventory and high content of Pu-240
- Difficulty to process fuel (e.g. TRISO)
- High spent fuel radiation barriers
- Low ratio of fissile to fuel volume, both in compact and pebble type

A relevant PR characteristic of pebble bed reactors relative to non-proliferation is the huge number of pebbles that would have to be diverted to accumulate a Significant Quantity of plutonium for weapons production. This poses also difficulty in nuclear material accountancy and control, but remains a considerable advantage over LWRs, in which this same amount of plutonium could be found in two spent fuel assemblies.

The high proliferation resistance is primarily due to the refractory coated fuel form and the low fissile fuel volume fraction. The refractory coatings provide a containment from which it is difficult to retrieve fissile materials. The technology for the reprocessing of the TRISO fuel has not been fully established yet, providing therefore a considerable increase of PR. In fact, PUREX cannot be directly applied to TRISO particles because silicon carbide coating layers are not dissolved by acid mixtures, and therefore require mechanical treatment to be removed.

GCR systems are able to reach very high burn-ups, which are far beyond the possibilities offered by other thermal reactors (with the exception of molten salt reactors). In particular the GT-MHR with optimized TRISO kernels prismatic fuel in graphite channels, could achieve a deep-burn capability allowing for essentially complete plutonium 239 fission and transmutation of 90% of all TRU waste in a single burn-up, minimizing the proliferation risk in the use of this fuel form, as well as limiting the generation of secondary waste [15].

As for GFR, the reactor system is optimized to recycle low-grade plutonium and minor actinides. Spent fuel treatment and re-fabrication will have to be performed in extensively monitored hot cells, which would facilitate the application of safeguard measures. With a high radioactive barrier, the fuel recycled in the system would be quite unattractive and difficult to handle for proliferation activities.

## 7.5. PP Intrinsic features

In terms of passive safety in case of loss-of-coolant events, GCR present the following characteristics:

- Helium coolant, which is single phase, inert, and has no reactivity effects
- For GT-MHR, graphite core with high heat capacity and structural stability at very high temperatures
- Refractory coated particle fuel, retaining fission products up to temperatures much higher than normal operation
- Negative temperature coefficient of reactivity, which inherently shuts down the core above normal operating temperatures
- Low power density core in steel reactor vessel surrounded by a natural circulation reactor cavity cooling system, as in GT-MHR
- Removal of decay heat by heat conduction, thermal radiation, and natural convection keeping fuel particle temperatures below damage limits

# 8. Liquid Metal Cooled Reactor Systems

Thirty-three liquid metal concept descriptions, from eight different countries, were considered by GIF's TWG3 [12]. Only four concepts of the 33 LMCRs were selected after the first screening round, and two (Sodium cooled and Lead/Lead-Bismuth cooled) eventually resulted as reference concepts.

## 8.1. Fuels and Fuel re-processing

All design concepts foresee fast neutron spectrum for efficient conversion of fertile uranium and a closed fuel cycle, with full actinide recycle fuel cycle at either central or regional fuel cycle facilities. The options for fuel type are still under consideration with the various design.

### 8.1.1. Mixed-Oxide Fuel

MOX ($PuO_2$, $UO_2$) has been the reference fast reactor fuel for over 20 years, with the demonstration of high burn-up mixed oxide fuel achieved in the FFTF (USA), PHENIX (France), MONJU (Japan), and PFR (UK). MOX recycling options include the advanced aqueous process, with uranium and plutonium co-extraction, along with most of the minor actinides (see par. 5.1).

### 8.1.2. Metal fuel

Metal fuels were reconsidered in the 80's, and haven't so far been tested comparably long to MOX, reaching lower burn-ups, i.e. up to 10% FIMA compared to more than 20% FIMA for MOX. Examples of metal fuels are ternary metallic alloys U-Pu-Zr or U-TRU-Zr.
Characteristics of metal bonded fuel are its higher density, yielding a harder neutron spectrum than oxide fuel and a smaller core volume, and high thermal conductivity, reducing the operating temperature of the fuel. Re-processing is done by pyro-metallurgical processes (see par. 5.2).

### 8.1.3. Nitride fuels

The state of development of nitride fuels is modest when compared to either the mixed oxide or the metal alloy. They are attractive for their high heavy metal density, good thermal conductivity and excellent compatibility with sodium and lead. But the amount of testing is still very small.

## 8.2. GEN-IV Sodium-Cooled Fast Reactor System (SFR)

For its density, heat transfer characteristics, and compatibility with the stainless steel materials of construction, sodium remains the coolant chosen in most fast reactor design development. Sodium cooled fast reactors are primary candidates for nearest term development in countries like France, within a European collaborative framework, USA (as Advanced Burner Reactor foreseen by Global Nuclear Energy Partnership (GNEP) by 2025 [18]), as well as India and China.

Sodium has however two major disadvantages: its chemical reactivity, which has caused problems in the past in French and Japanese reactors, and its positive void coefficient of reactivity in most plutonium-fuelled applications, which needs to be further researched upon to eliminate possible safety issues.

Two design options have been selected by GIF for further consideration, with different power ranges:

Medium power (150-500 MWe), characterized by:

- Metal uranium-plutonium-minor-actinide-zirconium alloy fuel and
- Pyrometallurgical processing in co-located facilities.
  Examples are KALIMER and JSFR designs.

Large power (500-1500 MWe), characterized by:

- MOX uranium-plutonium fuel and
- Advanced aqueous processing, possibly at a central location supporting several reactors.

The latter is the reference concept for the EISOFAR Specific Support Action (*Roadmap for a European Innovative Sodium cooled Fast Reactor* , EURATOM FP6) [19].

### 8.3. GEN-IV Lead-Cooled Fast Reactor System (LFR)

The design options for lead or lead-bismuth eutectic liquid-metal-cooled reactor system encompass various sizes:

- Battery of 50-150 MWe units, with very long refueling interval, factory made.
- Module of 300-400 MWe
- Large monolithic plant at 1200 MWe

The fuel options considered are metal or nitride-based, containing fertile uranium and transuranics. Lead or lead-bismuth eutectic offer important attributes as a fast reactor coolant. They are indeed neutronically superior to other liquid metal coolants (i.e. capture less than sodium and are less activated), they are inert, and they have very high boiling temperature and low vapour pressure. The total core void reactivity coefficient is negative.

Use of lead or lead-bismuth coolants, however, raises some safety and reliability concerns, such as the corrosion of the structural materials and the production of volatile and radioactive Po-210.

The experience gained with Pb-Bi eutectic cooled reactors in the Russian nuclear submarines, however indicates that many of the technical problems are overcome with adequate design and manufacture.

An example of innovative SMR design proposed is the Russian RBEC-M, cooled by lead flowing by natural convection and gas lift, sized between 120-400 MWth.

### 8.4. PR Intrinsic features

The intrinsic proliferation resistance features of LMCR can be summarized as:

- High burn-up and hence high spent fuel radiological barrier (up to 150 GWd/tHM)
- Inherently low decontamination factor of fuel
- Non-aqueous or advanced aqueous fuel reprocessing methods, with incomplete removal of fission products and MA
- No separation of uranium and plutonium at any fuel cycle stage, with either reprocessing method

The most PR resistant design seems to be the Pb/Bi battery, with a small size core with a very long core life attaining up to 30 years (typically 10 years). The reactor module is factory made, and shipped to the plant site. It would require little action from the operators, who have no access to the fuel, possible only in the factory.

## 9. Non-Conventional Reactor Systems

32 non-conventional innovative energy systems were considered by GIF's TWG4 [13], encompassing a wide variety of coolants and fuel designs.

The largest groups of NCRs are Liquid Core, Molten Salts and Gas Core ones, besides other concepts like Direct Energy Conversion and Waste Minimization.

Of the 32 initial ones, only one Liquid Core Reactor System and two types of Molten Salt Reactors also considered in IAEA TECDOC's list were selected for GIF's second screening, i.e. molten salt core and molten salt cooled ones. Only the former was eventually selected as GEN-IV reference.

## 9.1. GEN-IV Molten Salt Core Reactor (MSR)

Initially developed in the early 1950s in the USA, when a small test reactor was successfully operated, this design has the potential to be a thermal $Th^{232}$ -$U^{233}$ breeder power reactor with high thermal efficiency [13].

The reference power level is 1000 MWe, operating at low pressure (5 bar) with a coolant outlet temperature of 700 $^{o}$C, allowing high thermal efficiency and potential for hydrogen production.

The heat generated in the molten salt is transferred to a secondary coolant system through an intermediate heat exchanger, and then to the power conversion system, which can be a high-temperature steam cycle or possibly a helium gas turbine cycle.

The MSR is characterized by a thermal to epithermal neutron spectrum and a closed fuel cycle for the efficient utilization of plutonium and minor actinides, with full actinide recycle fuel.
There are four fuel cycle options. Actinide burning with continuous recycling id the GEN-IV reference option.

The fuel is liquid, i.e. a circulating mixture of sodium, zirconium and uranium fluorides, hence there is no need for fuel fabrication. Molten salt flows up through vertical unclad graphite core channels. Besides operating as a thermal breeder reactor on a $Th^{232}$ -$U^{233}$ fuel cycle, with very low resource demands, the MSR could be loaded with both Th and U-238.

It is claimed that the production of 1 TWh would require 100 kg of natural thorium for an MSR, instead of 20 tons of natural uranium with a PWR. The MSR hence minimizes the waste output [13]. The development of this concept is indicatively foreseen by 2025.

## 9.2. PR Intrinsic features

A molten salt core reactor does not require fuel fabrication, which is a very expensive and difficult process for fuel including the higher actinides (americium and curium). Actinides and most fission products are directly formed, or added, in the liquid molten salt coolant and completely burnt. The thorium fuel cycle is foreseen to greatly minimize the generation of higher actinides, compared to an LWR [13].

The dilute concentration of actinides in the salt avoids the handling of concentrated higher actinides with their very high decay-heat generation rates. These aspects would seem to lower the technological barrier to proliferation, however there is no recycling of actinides out of a MSR once they are added to the molten salt, and the inventory of actinides required to maintain a critical MSR is low compared to other reactors.

The combined reactor and fuel cycle fissile inventory is low, because MSR is a thermal neutron reactor with a small critical mass, and with high power density in the liquid. Moreover, it is claimed that MSR's fissile material isotopic composition would be unfavourable for use in weapons fabrication.

Due to the scarce studies to date, the non-proliferation characteristics of a MSR are conservatively defined as equivalent to an LWR. However, given the radically different characteristics of the MSR, there is ground for an effective significant higher proliferation resistance which needs to be better investigated.

## 10. The six GEN-IV reference concepts summary

Table 2 summarises the main technical characteristics of the six reference concepts, with their options.

| | Size [MWe] | Spectrum | Fuel | T out [°C] | Uses | Due by |
|---|---|---|---|---|---|---|
| GFR | 288 | Fast | Fertile U, Actinide-carbides or nitrides, ceramic clad, or ceramic composite | 850 | Electricity, hydrogen | 2025 |
| LFR | ▪ 50-150<br>▪ 300-400<br>▪ 1200 | Fast | Fertile U, TRU, nitride or metallic | 550-800 | Electricity, hydrogen | 2025 |
| MSR | 1000 | Thermal/ Epitherm. | Liquid mixture MA, U, FP, Na, Zr | 700-850 | Electricity, hydrogen | 2025 |
| SFR | 1) 150 - 500<br>2) 500 -1500 | Fast | 1) U-Pu-MA-Zr metal<br>2) MOX (and MA) | 550 | Electricity | 2015 |
| SCWR | 1500 | Thermal; Fast | U oxides | 550 | Electricity | 2025 |
| VHTR | 600 MWth | Thermal | ZrC coated particles in blocks, pins or pebbles | 1000 | Hydrogen, heat, (electricity) | 2020 |

**Table 2**: Summary of main technical characteristics.

## 11. Considerations on intrinsic PP features of innovative nuclear systems

As recalled in par. 3, GEN-IV Roadmap assessment considered Physical Protection according to *Vulnerability of Installations* criterion, and *Passive safety Features* metric.

The main scope of intrinsic physical protection is to limit the consequences of attacks finalised at theft of nuclear material or sabotage. Extrinsic security measures can be adopted to protect the installation from both threats, whereas the design can rely on intrinsic physical protection features to improve the response to damage caused by sabotage.

In response to sabotage, a NES should exhibit a high capability to withstand lack of coolant and consequent very severe temperature rise without core meltdown. To this scope, refractory materials with high heat capacity and temperature resistance are a useful first passive structural feature, largely foreseen as we saw in gas-cooled reactor systems.

Decay heat removal should be possible without intervention of active systems, i.e. as much as possible with passive functions, without requirements of AC power. Cooling by natural convection as in LFR is a suitable option to this scope. Low-pressure fluids are preferable to reduce safety issues in case of circuit break. These features are typical of liquid-metal-cooled reactor systems.

The location of nuclear system components should be in structures able to withstand attacks by means of energetic projectiles.

As much as possible, implementation of highly automated control systems is considered beneficial to safety, and provides resistance to sabotage, reducing need for human intervention, with proper cross-checking of alarm parameters.

Based on the above, most of the 19 innovative NES concepts first selected by GEN-IV Roadmap where found to be similar of slightly better than the reference Advanced LWR, In particular PBMR, GT-

MHR, SFR with Metal fuel were considered better than ALWR for their high thermal inertia, heat removal means and level of passive safety features.

## 12. Considerations on intrinsic PR features of innovative nuclear systems

According to the PR metrics considered by Roadmap, the innovative NES designs could roughly be grouped according to the following intrinsic PR characteristics:

- Long-life not-accessible core,
- Once-through high-burn-up fuel not recycled
- High Burn-up spent fuel with pyro-processing,
- Low Burn-up spent fuel with aqueous reprocessing

In particular, as we saw, we can identify some intrinsic features adopted by GEN-IV innovative NES reference designs to increase proliferation resistance:

- Open cycle and deep burn spent fuel, with no available technology for re-processing ceramic composite fuel element (VHTR)
- Modular factory-made cassette core inaccessible by the operator (LFR battery option)
- Complete burning and no recycling of actinides in molten salts (MSR)
- MOX or metal-bonded fuel charged with Minor Actinides (SFR)

The design choices made to pursue the sustainability, economics, safety and reliability, and proliferation Resistance and physical Protection goals are all interdependent. In some cases apparent benefits also have unexpected drawbacks. This is true also for PR and PP, as we summarise hereinafter.

### 12.1. Fuel and Actinide Management

Most of the innovative NES designs have a fast neutron spectrum primarily for burning Minor Actinides (Am, Np, Cm) and for breeding fissile material to improve the system sustainability.

The goal of this type of design is that of ensuring a self-sustainable operation or breed fuel to feed other reactors, and possibly reduce the need of fuel enrichment and related enrichment facilities, thus enhancing proliferation resistance.

To be intrinsically more proliferation resistant, and avoid diversion of separated plutonium, NES designs should foresee core concepts without fertile blankets. The created plutonium should be within the fissile fuel, and recycled by co-extraction with other actinides, without separation (see par. 5). Alternatively, radial blankets fertile charged with minors actinides with purposes of transmutation (and reduction of attractiveness) could be considered.

### 12.2. Burn-up

Intrinsic radiological barriers are certainly enhanced by longer irradiation and higher burn-up. Fuel deep burn not only increases the specific energy production, but also degrades the plutonium isotopic composition and creates a higher radiological barrier to its handling, hence a higher technological difficulty for the proliferation actor. However fuel handing becomes more complicated also for the operator, increasing costs and affecting economics.

In this respect, we should add that literature plutonium generation calculation as function of burn-up show that Pu-239 build-up with irradiation tends to slow down after 33 GWd/tHM, whereas the production of higher isotopes (and Pu-238) continues to increase nearly linearly. For burn-up around 60 GWd/tHM, isotope 239 percentage decreases from 60% to 50% of the total plutonium content [20], [21]. Further increase in burn-up would only slightly degrade plutonium isotopics.

Higher burn-up reduces the frequency of fresh and spent-fuel handling and spent-fuel transportation requirements. By reducing spent-fuel inventories, it also reduces the total mass of associated plutonium, which anyway still remains substantial, and the need for on-site storage of spent fuel [22].

### 12.3. Facilities location

As for the location of recycle facilities, various options with PR and PP advantages and drawbacks are under consideration. The integration of the fuel cycle in the nuclear site would minimize both transports of nuclear materials (restricted to make-up fuel) and the total amount of nuclear materials needed for the operation of the system over its entire life, as the fissile fuel needed is bred in situ from fertile fuel. However application of safeguards to many reprocessing plants would become more complicated.

Regional or centralized reprocessing facilities would instead facilitate centralized application of safeguards, but require nuclear material transportation at various distances, hence posing physical protection issues.

The attractiveness of the type of feed fuel is another factor to be considered both from PR and PP points of view: besides the isotopic composition of spent fuel, with its radiological barrier, also fresh material could be a possible proliferation target, especially MOX fuel, which is considered direct use by IAEA.

## Acronyms

| | |
|---|---|
| AHWR | Advanced Heavy Water Reactor |
| ALWR | Advanced Light Water Reactor |
| FIMA | Fraction of Initial Metallic Atoms |
| FFTF | Fast Flux Test Facility (Hanford site, U.S.) |
| GCR | Gas-Cooled Reactor |
| GFR | Gas-cooled Fast Reactor |
| GNEP | Global Nuclear Energy Partnership |
| GT-MHR | Gas Turbine .Modular Helium Reactor |
| HEU | High-Enriched Uranium |
| HTTR | High Temperature engineering Test Reactor |
| IMR | Integrated Modular water Reactor |
| IRIS | International Reactor Innovative and Secure |
| JSFR | Japan Sodium fast Reactor |
| KALIMER | Korea Advanced Liquid Metal Reactor |
| LEU | Low-Enriched Uranium |
| LMFBR | Liquid Metal Fast Breeder Reactor (an archaic term for fast reactors) |
| LMR | Liquid Metal Reactor |
| LOCA | Loss Of Coolant Accident |
| LWR | Light Water Reactor |
| MARS | Multipurpose Advanced Reactor, inherently Safe |
| MOX | Mixed uranium-plutonium OXide (fuel) |
| MSR | Molten Salt Reactor |
| PBMR | Pebble Bed Modular Reactor (in South Africa) |
| PBR | Pebble Bed Reactor |
| PFR | Prototype Fast Reactor. |
| PMR | Prismatic fuel Modular Reactor |
| PRISM | Power Reactor, Inherently Safe Module |
| PWR | Pressurized Water Reactor |
| RBEC-M | Lead-Bismuth Cooled Reactor with high level of natural circulation |
| RMWR | Reduced Moderation Water Reactor |

| | |
|---|---|
| SMART | System-integrated Modular Advanced Reactor |
| TRISO | refractory (coated particle fuel) |
| TRU | Transuranic |
| TWG | Technical Working Group |
| VBER-300 | Water Cooled Modular Power Reactor |
| VHTR | Very-High-Temperature Reactor |

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# Session 4

# European Support programmes

# The Provision of Research, Development and Implementation Support for the IAEA's Department of Safeguards – What next?

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***Abstract:***

*2007 is the 50th Anniversary of the IAEA. For the last 30 years the IAEA has relied on Member States to both provide and fund safeguards specific development and implementation support work. Most of this support is channelled through the 20 Member State Support Programmes.*

*This paper will describe the current system of support with 20 Member State Support Programmes with over 200 projects spending over US$20 million/year. It will identify some challenges for the current system, describe some successful projects and make recommendations.*

*The paper will recommend that it is timely to examine possible ways to improve the current system of collaboration with technology holders. The objective is to stimulate debate on the mechanisms for longer term commitments of support for research and development in international safeguards, particularly in the area of new technology. The importance of sustained cooperative programmes will be emphasized.*

**Keywords:** IAEA; Member State Support Programmes; R&D

**Theme:** New verification technologies

**Topic:** Research and Development for Safeguards – A new approach?

## 1. A Brief History of the Member State Support Programmes

Since the 1970s research, development and implementation support work in the Department of Safeguards has been assisted by Member State Support Programmes (MSSPs). These MSSPs voluntarily fund work to strengthen international safeguards through improvements in the effectiveness and efficiency of international safeguards by transferring technology and expertise from Member States to the IAEA.

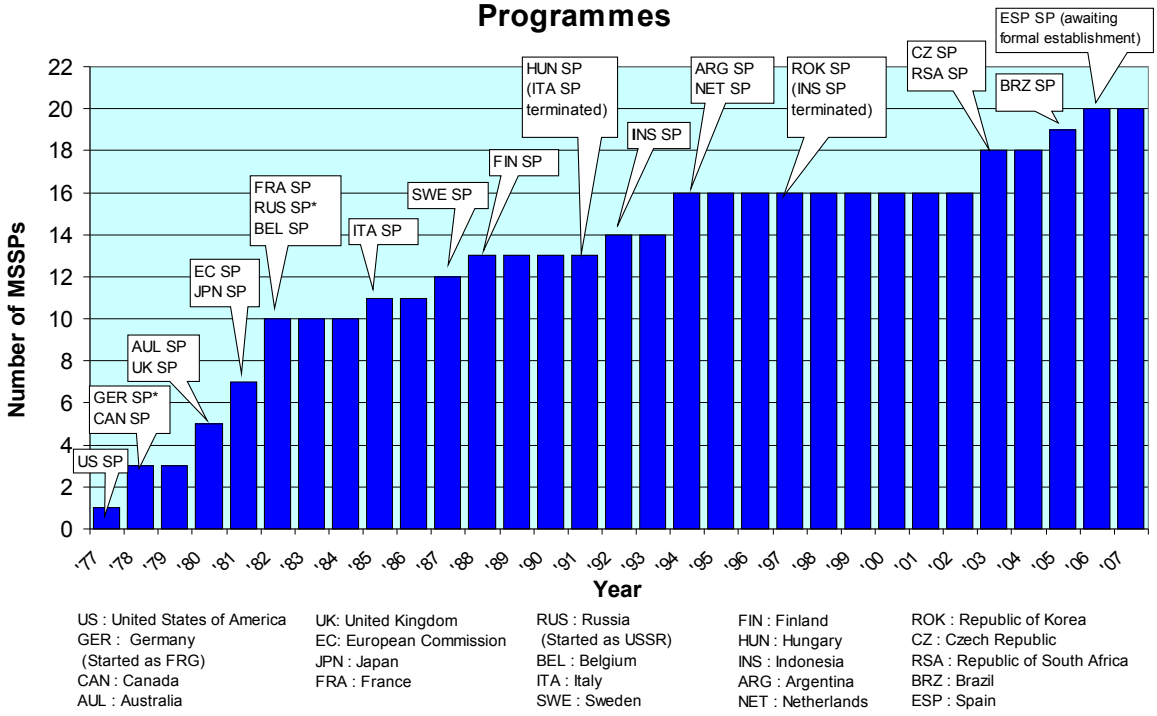
MSSPs are required because the Department of Safeguards has neither the resources nor the required infrastructure to maintain the comprehensive research, development and implementation programme that will enable it to fulfil its verification mandate particularly as stated in the IAEA's Medium Term Strategy for 2006-2011, (IAEA Gov/2005/8: Objective C.1: Provide credible assurances to the international community that States are honouring their safeguards obligations).

MSSP resources are used to support the development of safeguards equipment, to develop safeguards specific information technology software and hardware, and to develop new safeguards concepts, approaches, methodologies and techniques. The work also supports the analysis of nuclear materials and environmental samples. Additionally a large component of the support involves training activities for both Agency staff and staff from SSACs.

The first MSSPs were established in the 1970s (Figure 1). There are now 20 MSSPs. These are;

Argentina, Australia, Belgium, Brazil, Canada, the Czech Republic, the European Commission, Finland, France, Germany, Hungary, Japan, the Netherlands, the Republic of South Africa, the Republic of Korea, The Russian Federation, Spain, Sweden, the United Kingdom and the United States of America.

**Figure 1. Starting Dates of Member State Support Programmes**




The basis of the current system can be found in paper submitted by Thorstensen and Larrimore to de Clerk and the Geneva Sub-Group on Safeguards in 1989. At that time there were over 200 tasks being supported by 13 support programmes contributing an estimated US$12-14 million. This paper established the support programmes organization within the IAEA by assigning roles to the Support Programmes Administration, Division Directors, Task Officers and Specialists. The paper introduces

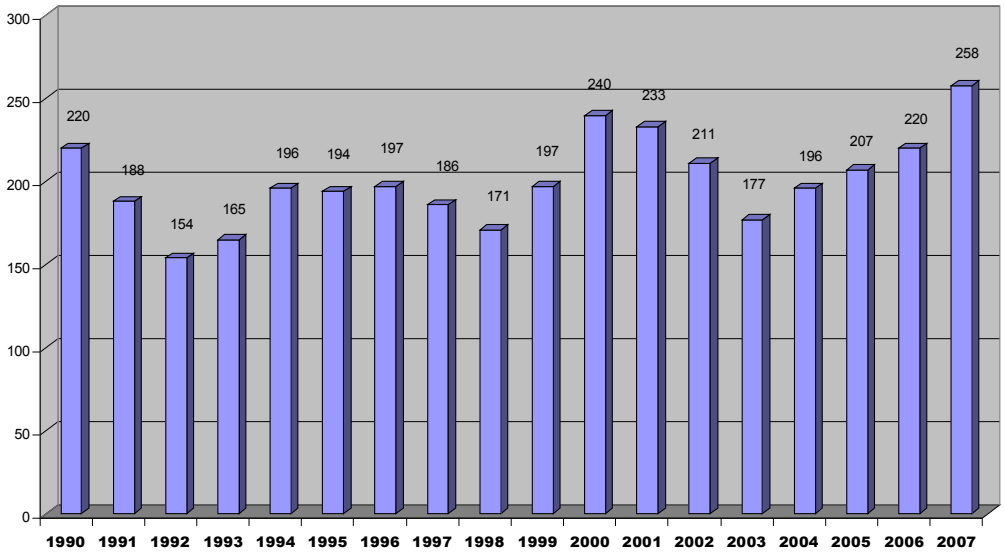
the task proposal system (SP-1), task agreements (SP-2s) and a new Support Programme Information and Communication system (SPR-ICS).

This system operated largely unchanged until 2000 when, as a result of a number of evaluations, the Agency initiated improvements, including a new system for the approval of support programme task proposals, a project management system for all MSSP tasks and the transfer of responsibility for the Department's R&D Programme to the Support Programmes Administration. The implementation and early success of this new system is described in a paper submitted to the London ESARDA in 2005 (Ref 1).

In 2006 it is estimated that MSSPs contributed, in cash and in kind, over US$20 million to the Department of Safeguards. This level of contribution has slowly increased with time and still remains a significant resource for a Department whose regular budget just exceeds US$120 million.

Currently there are a growing number of tasks with the total number standing at 258 tasks (Figure 2).

**Figure 2. Total number of tasks at the beginning of the year**
**(excluding Cost Free Expert tasks)**




This paper describes the need for further improvements to the support programmes mechanism based on changing legal and technical environments. It suggests a way to progress based on successes with the current system that will provide cost effective and efficient use of MSSP funds.

## 2. The Need to Change

There are a number of factors that need to be accommodated by changes to the MSSP mechanism.

1. Legal and International Public Sector Accounting Standards

2. Experience with successful projects with the current mechanism such as

- The next generation surveillance system project
- The seals project
- The satellite imagery analysis project

3. Equipment development life cycle and escalating costs

They are discussed below in detail.

## Legal and International Public Sector Accounting Standards

The MSSP mechanism has always been regarded as a “low maintenance” system whereby expertise and technology have been transferred to the Agency with the minimum of bureaucracy. As Member States increasing require more from the Agency not just in terms of its mission but also in reporting and transparency an increasing number of initiatives are contributing to increased bureaucracy.

The Agency has appropriate rules governing the gift of money, services, equipment and facilities. The Rules, set out in INFCIRC/370/Rev2, concerning (a) gifts of services, equipment and facilities ("in kind" contributions) and (b) voluntary contributions of money, were adopted by the Board of Governors and the General Conference respectively. They are intended to ensure that all gifts and contributions to the Agency, from whatever source, can be incorporated into approved activities, will not involve the Agency in additional expense and are not accompanied with conditions that will hamper the Agency's activities.

The authority to accept these gifts has been delegated to the Deputy Director General for Management. In order to discharge his responsibilities a system has been put in place that requires each Deputy Director General to certify to the Department of Management that each such “gift” complies with the above rules. This legal requirement places a significant administrative burden on the Department of Safeguards, the IAEA’s legal staff and the budget officers.

In addition to the above legal requirements it is now proposed that the Agency adopt International Public Sector Accounting Standards (IPSAS). These standards, which have been praised by the IAEA’s external auditor, have the benefit of *ipso facto* “improved transparency, comprehensive and consistent information about costs and income and better support for results based management”. With respect to voluntary contributions in cash, donors will receive statements that are consistent and comparable with those of other organizations that work to the same standard. With respect to in-kind voluntary contributions it is expected that a cash value will be placed on these by the Agency and the value reported in the IAEA’s accounts. Again this will become an additional administrative burden for the Agency and, unless an appropriate system is in place, a source of conflict with MSSPs should the value placed on the contribution be at odds with the amount spent.

## Project case histories

In order improve the MSSP mechanism it is important to recognize successful projects and to analyse the reason for that success. The following projects are being discussed because they are tangible achievements and illustrate certain arrangements. They are not the only successful projects being undertaken.

## The Next Generation Surveillance System (NGSS)

The development of the first generation of digital equipment systems was a learning experience for the IAEA and Member State Support Programmes. As well as the technical issues involved in the development of a replacement for the existing systems, a number of legal and procurement issues were encountered. It would be fair to say that that these issues presented as great a challenge for development as the technical issues and they caused many delays to the development programmes.

In setting up the project for the NGSS, the IAEA, in cooperation with the US and German Support Programmes, tried proactively to alleviate the effects of legal and procurement rules, as well as developing a technical project plan.

The detailed planning has recognized a development life-cycle of five years and an equipment replacement life-cycle of six years, with both Support Programmes and the Agency agreeing to long term arrangements i.e.; for eleven years. Add to this an expected lifetime of 10 years for such systems, then the assumption is that there will be continuing cooperation for at least the next 20 years on this single project. Any intellectual property rights issues and issues with respect to the manufacture and procurement of the developments have as much as possible been agreed in advance. For example, it is recognized that should the product of the development meet the IAEA's detailed user requirements, then sole source procurement action will be possible.

The NGSS project was initiated in 2002. At the time of drafting this paper, the project has completed the conceptual and detailed design phases. Prototype development has now started with the development project completion date expected to be November 2008.

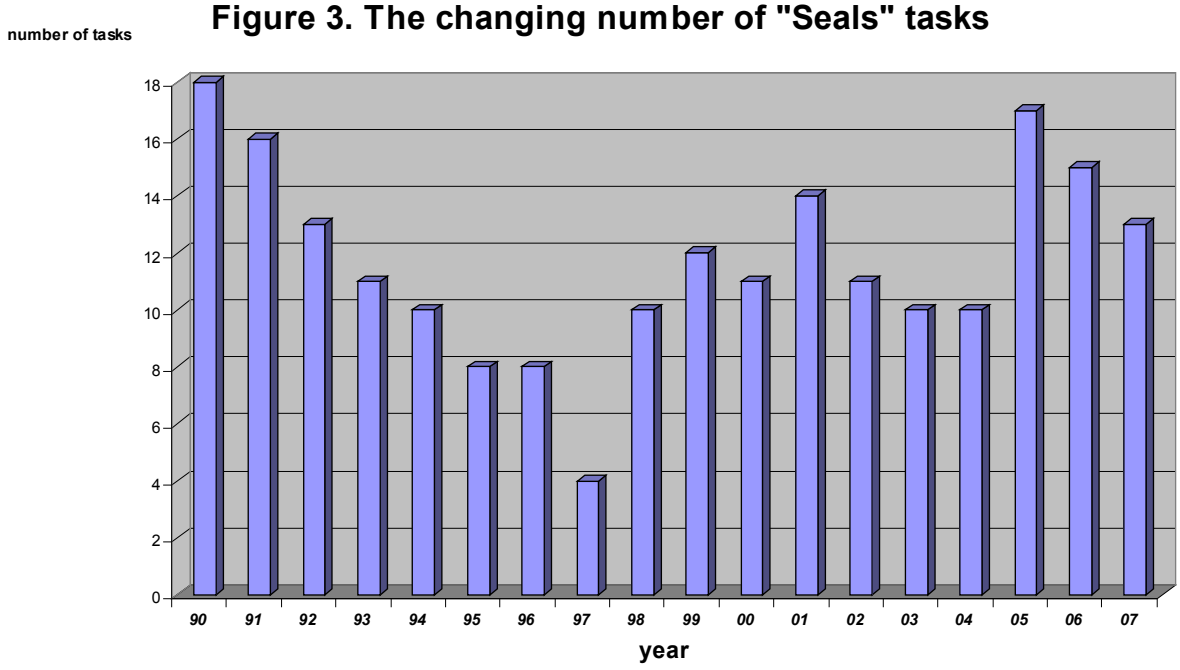
An important factor in the success of the project so far has been the building of a project team managed by the IAEA but including partners in the German Support Programme, the US Support Programme, the European Commission and a number of Departments within the IAEA including Legal and Procurement.

## The Improved Techniques and Instruments for Sealing and Containment Verifications ("Seals") Project

The use of the "Seals" project as a vehicle for discussion in the equipment development area is significant. The seals project is a focused project that is a microcosm of the other equipment development projects. The conclusions drawn within this project have direct applicability within other areas of equipment development.

The development work for seals has had a mixed history (Figure 3). It is one that has gone through a number of stages. There was competitive development in the 1990s where 3 electronic sealing projects were managed and implemented, there was a decline in the work performed in the late 1990s and there has been a cooperative rebuilding exercise incorporating coordinated technical meetings in 2004 and 2007. In particular since 2000 the project has become focused and forward looking.

**Figure 3. The changing number of "Seals" tasks**



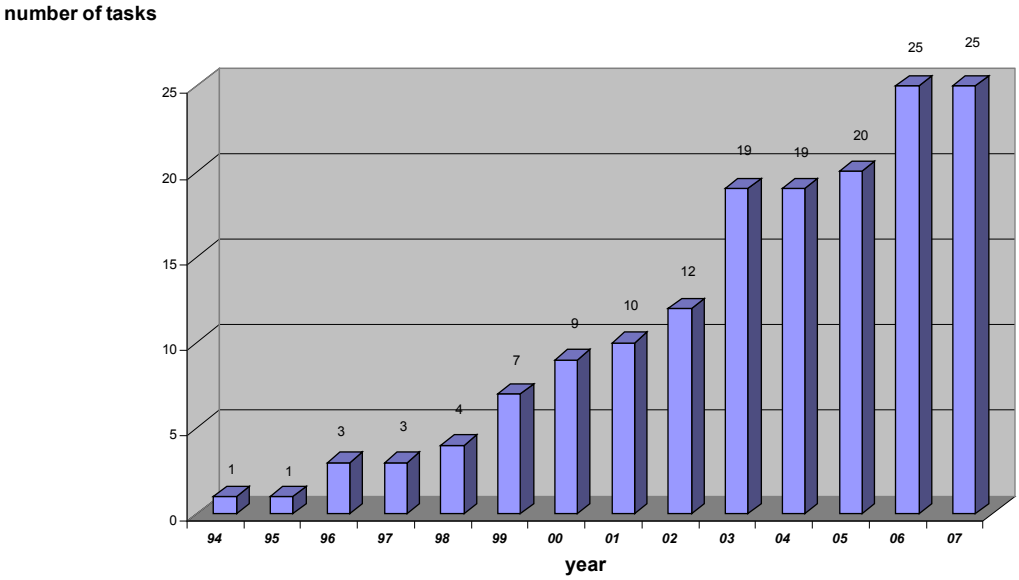

The 2004 technical meeting developed a strategy and programme and the increase in MSSP tasks as a result of this meeting can be clearly seen in the above chart. The subsequent review of this programme in 2007 has shown progress with the tasks but has also made general conclusions about how the work can continue. The first conclusion is that the technical community requires detailed user requirements from the IAEA in order to increase understanding of the Agency's specific needs and also no single technology can solve all the Agency's needs. Finally the IAEA does not have the resources to manage all the R&D work. It is suggested that cooperation is improved within the technical community and it requests MSSPs to act as the Agency's advocates identifying government and private organizations and can assist. It is strongly believed that both the IAEA and MSSPs do not have the resources for competitive developments.

## The Satellite Imagery Project

There is no doubt that the IAEA's capability to obtain and analyse satellite images would not be as mature had the development relied upon the regular budget. From the first MSSP joint task agreed with the UK and German Support Programmes in 1994 the Agency has introduced a significant new capability. This now includes a multi-million US$ imagery analysis laboratory with 10 high-powered imagery workstations supported by imagery exploitation and GIS software, enhanced security and significant storage capabilities.

The Satellite Imagery Analysis Unit consists of 13 staff, supported by 4 consultants. It has been supported by 13 MSSP consultants since 1999. The Unit still enjoys significant support from MSSPs with the number of tasks steadily increasing to 25 (Figure 4). The Agency attaches particular importance to the training of its own in-house resources and this has been a significant component of the support supplied by MSSPs.

**Figure 4. The Changing the number of "Satellite Imagery" tasks**

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Although a complex and wide-ranging technology, the need for work is well defined and easily understood by budget holders. The results of such work is clearly visible within the context of the IAEAs Mission.

## 3. What happens next?

### The “do nothing” option

If the current arrangements persist and the new legal and accounting practices are applied then the administration costs will increase and efficiency of the MSSP mechanism will decrease. The drive for greater transparency and accountability is an appropriate initiative but it comes at a cost. It must be recognized by MSSPs that the IAEA’s rules will not make an exception for MSSPs, therefore in order for the work to continue in an effective and efficient manner, the arrangements for giving and receiving support must adapt.

### The proactive approach

There are a number of initiatives that can be taken to further improve the MSSP mechanism. It must be stated that “one size” will not fit all and that different approaches will be needed for different projects and different MSSPs. Based on the above evidence of continued administrative complexity and the lessons learnt from the discussed projects, there are a number of initiatives that can be studied

- Establish long term arrangements
- Focused projects
- Associate laboratories and MSSP Projects
- Equipment development standards and sustainability

## Long Term Arrangements

The current position of annual pledge letters and pledge letters for individual contributions initiates a process within the Agency that is lengthy involving the drafting of a letter by the Department of Safeguards and negotiations with the IAEA's Office of Legal Affairs and the Division of Budget and Finance. Frequently the donor is involved in these negotiations and much care is taken to be accurate and precise. It is a process that can take months. It would lead to significant savings for both the IAEA and for MSSPs if a model agreement could be agreed and that this model agreement used to establish long term arrangements. Such an agreement has already been reached with the Canadian Safeguards Support Programme.

The establishment of such long term agreements would not only benefit the IAEA but it would also bring together donating organizations with the States' Ministries of Foreign Affairs. This would have an impact similar to that seen with Nuclear Security where funding seems to have been easier to justify and obtain.

It is recognized that the *de juro* case is that Member States can only provide annual funding but the *de facto* case, evidenced by 30 years of contributions, can only lead to the conclusion MSSPs are here to stay. It should be recognized that MSSPs pay for, *inter alia*, routine operations and that they are an integral part of the safeguards regime. Can anyone imagine the Member States of the IAEA paying for the IAEA to do its own R&D and to set up an R&D infrastructure?

## Recommendation 1

Member States with Safeguards Support Programmes should approach the IAEA in a coordinated manner to develop a long term model agreement.

## Focused Projects

As shown in the examples used, focused projects remain a successful vehicle for the provision of support. The IAEA, in partnership with MSSPs should examine whether the individual MSSP focused arrangements currently in place are the best for the provision of cooperative and successful projects. If the workshop approach for the seals project, the long term cooperative approach of the next generation surveillance system project and the focused approach for the satellite imagery project were to be combined perhaps a new structure of support could be developed. Perhaps to allow better focus the number of projects should be increased.

## Recommendation 2

More emphasis should be placed on project meetings involving technical staff.

## Recommendation 3

The IAEA should better describe each project in terms of tangible end results.

## "Associate" Laboratories and MSSP Projects

Given the current restrictions on budget and the expansion of the IAEA's mission, the major restriction on the amount of work undertaken through the MSSP mechanism is the IAEA's ability to manage the

tasks. As shown above, the number of tasks is already approaching the limit. Yet why should this limitation restrict the work being undertaken?

It is recognized that much safeguards related development work is ongoing without the involvement of the Agency. However, from an IAEA view-point, this work is not always closely related to its future priorities and it would benefit all round from an increased understanding of the IAEA's agenda. Whilst organizations such as the European Safeguards Research and Development Association and the Institute of Nuclear Materials Management add value to this process the contacts are largely in an *ad hoc* manner and by personal, rather than institutional approaches.

In addition to the projects mentioned in the case studies, the novel technologies project (Ref 2) now seems likely to yield substantive results. It is possible that the detailed implementation of such technologies will require an infrastructure similar to that already existing for more traditional technologies. It is difficult to see this developing further without MSSP initiatives of a scale similar to that shown for the use of commercial satellite imagery.

### Recommendation 4

New institutional support for the safeguards should be developed. This may take the form of associate laboratories that manage multinational safeguards projects with the IAEA acting in a consultancy role. This support could begin by identifying a small number of suitable laboratories with a geographical distribution that could lead developments in "novel technologies". A starting point could be the establishment of an "associate" laboratory to drive the emerging laser based applications through to implementation. Such a model could then be applied to other projects.

### Equipment Development

Such a complex topic as equipment development is the subject of a separate paper, however, it can be seen from the seals project that work in this area is characterised by expensive development costs for a narrow application and a small unit volume. This generally pushes the IAEA towards small to medium sized companies causing concerns about long term support. This is further illustrated by increased problems with component supply and the requirement to regularly update software. As well as supporting the argument for long term arrangements with MSSPs, these factors suggest that a new process be found to reduce the spiraling costs of equipment for safeguards use. Perhaps the use of "turn-key" projects should be investigated?

### Recommendation 5

Since Member States with MSSPs remain the principle suppliers of technology for safeguards, this paper recommends that a cooperative project be agreed. To facilitate this process the IAEA should draft a task proposal establishing a joint project to develop a new approach to safeguards equipment development.

## 4. Conclusions

MSSPs remain an essential part of the international safeguards regime and their resources will be required for the longer term. As shown in the paper the current programme of work is healthy and is producing results, however, the amount of work is nearing the maximum number of tasks the IAEA can manage. The paper recommends that action be taken now to reduce bureaucracy, focus projects and study new ways to support the IAEA. It is suggested that these new arrangements can be confirmed at the next MSSP Coordinators Meeting in 2008.

## 5. References

1. The Role of Member State Support Programmes in the Development of Technologies for the IAEA's Department of Safeguards, N.Khlebnikov, A.Hamilton, presented to ESARDA London 2005
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# Safeguards in France : much more than theory

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## Abstract:

Although there are already a lot of European commission reports related to the implementation of Euratom safeguards (including in France) and also IAEA's reports listing facilities under safeguards all around the world, it seems that there is no document describing the scope of these safeguards in France and in particular the way they are perceived by the French authorities.

In addition, questions are often asked by institutional partners of France about the national organisation allowing the follow-up of safeguards in France.

In order to answer these questions, the paper will aim to review how safeguards are actually implemented in France. In addition, the French views on the relevance, the advantages and constraints of these safeguards will be presented. Furthermore, the paper will briefly review the legally binding instruments for France.

The paper will also describe the French organisational scheme in the field of safeguards which originality is to associate complementary national organisations able to achieve a global follow up of safeguards implementation either by the European Commission or the IAEA taking into account political and technical aspects.

Finally, the paper will focus on complementary commitments made by France in the safeguards field through the French support program to IAEA safeguards. The aim of this program is to give the opportunity to the IAEA, and more broadly to the international community, to benefit from a technical knowledge and expertise at every stage of the fuel cycle as a result of the development of the French nuclear industry for several decades. The French support program will celebrate its 25$^{th}$ anniversary in 2007.

**Keywords:** safeguards, Euratom, IAEA, French organization, French support program

## Introduction

The purpose of the present document is to review the implementation of international safeguards on nuclear materials held in France and present the French organisation that is set up to ensure these controls are monitored. The French support programme to IAEA safeguards, which in 2007 celebrates 25 years existence, will also be described.

## 1. France, one of the most heavily controlled countries in the world

- **Typology of controls in France**

Nuclear materials in France are the subject of a wide variety of controls exercised by

- the operators
- the national authority
- the European Commission
- the International Atomic Energy Agency (IAEA).

These different controls do not have the same scope nor the same finality.

**. Operator's controls**

The operator is responsible for proper management of nuclear materials and implements internal procedures to ensure in particular that national and international regulations are being applied.

**. National controls**

National control is exercised over all materials for civil use and meets objectives essentially concerned with security and physical protection. It sets strict rules for nuclear materials accounting. National inspectors have the job of carrying out missions in the field. National control is placed under the responsibility of the High representative for defence in the Ministry of Industry.

**. Euratom safeguards**

Legal aspect

The “Euratom” control is exercised by the European Commission over all civil nuclear materials. It is an outcome of the 1957 Euratom treaty whose section VII provides for the enforcement of controls on nuclear materials.

Within the provisions of article 77 of this treaty, the commission must verify:

“- ores, source materials, and special fissile materials are not diverted from their intended use as declared by their users

- the provisions relating to supply and any particular safeguarding obligations assumed by the Community under an agreement concluded with a third State or an international organisation are complied with”.

The European Commission controls are exercised on the basis of Commission regulation No.302/05 of 8th Feb, 2005 relating to the application of Euratom safeguards (JOUE of 28/02/2005). An independent body of inspectors has the job of conducting inspections in various installations in the European Union.

Practical aspects

Checks conducted by the European Commission in France are significant for several reasons. For a start, France must supply the European Commission with information on all installations holding civil nuclear materials. Next, accounting data for these materials are regularly sent to the European Commission. Last, inspections are regularly conducted by the European Commission, making France the most heavily controlled country in the European Union along with the United Kingdom.

A few figures will give you a better idea of the extent of these controls. In 2006, 168 material balance areas came under inspection and 612 installations were declared as small holders. 261 inspections were carried out corresponding to 1233 person days of inspection.
These inspections were divided as follows: 86% for AREVA, 7% for EDF, 6% for the CEA, and 1% for others French actors involved in the nuclear field.

. **IAEA safeguards**

Legal aspects

IAEA safeguards concern materials that have been subjected to safeguards via specific legal commitments (France/third-State agreements, Euratom/third-State agreements etc.).

The controls are carried out on the basis of legal instruments signed and ratified by France, such as the safeguards agreement established on a trilateral basis between France, Euratom, and the IAEA which entered into force in 1981. This agreement is published under the IAEA reference "infcirc 290".

On 30th April, 2004 the additional protocol to France/Euratom/AIEA agreement entered into force at the same time as those of its European partners, on the basis of ratification act No. 2003-375 of 24th April, 2003.

As a reminder, France has been a signatory to the Non-proliferation Treaty (NPT) since 1992 and to the Comprehensive Nuclear Test Ban treaty since 1998.

Practical aspects

The scope of controls exercised by the IAEA is different from that of the European Commission. The IAEA can have access to materials submitted by France to its safeguards in two ways. First, it regularly receives accounting information concerning the materials under safeguards. Next, the Agency can verify these materials as soon as it designates an installation where the inspections will take place. The IAEA has until present concentrated its efforts on the inspection of reprocessing operations (La Hague plant) and on the manufacture and shipment of Mox (Melox and La Hague) to Non nuclear weapon states Five inspections took place in 2006, essentially at La Hague, corresponding to fifty person days or so of inspections. In the future, inspections are to be carried out at the centrifuge enrichment plant (GB2) in Pierrelatte in the South of France.

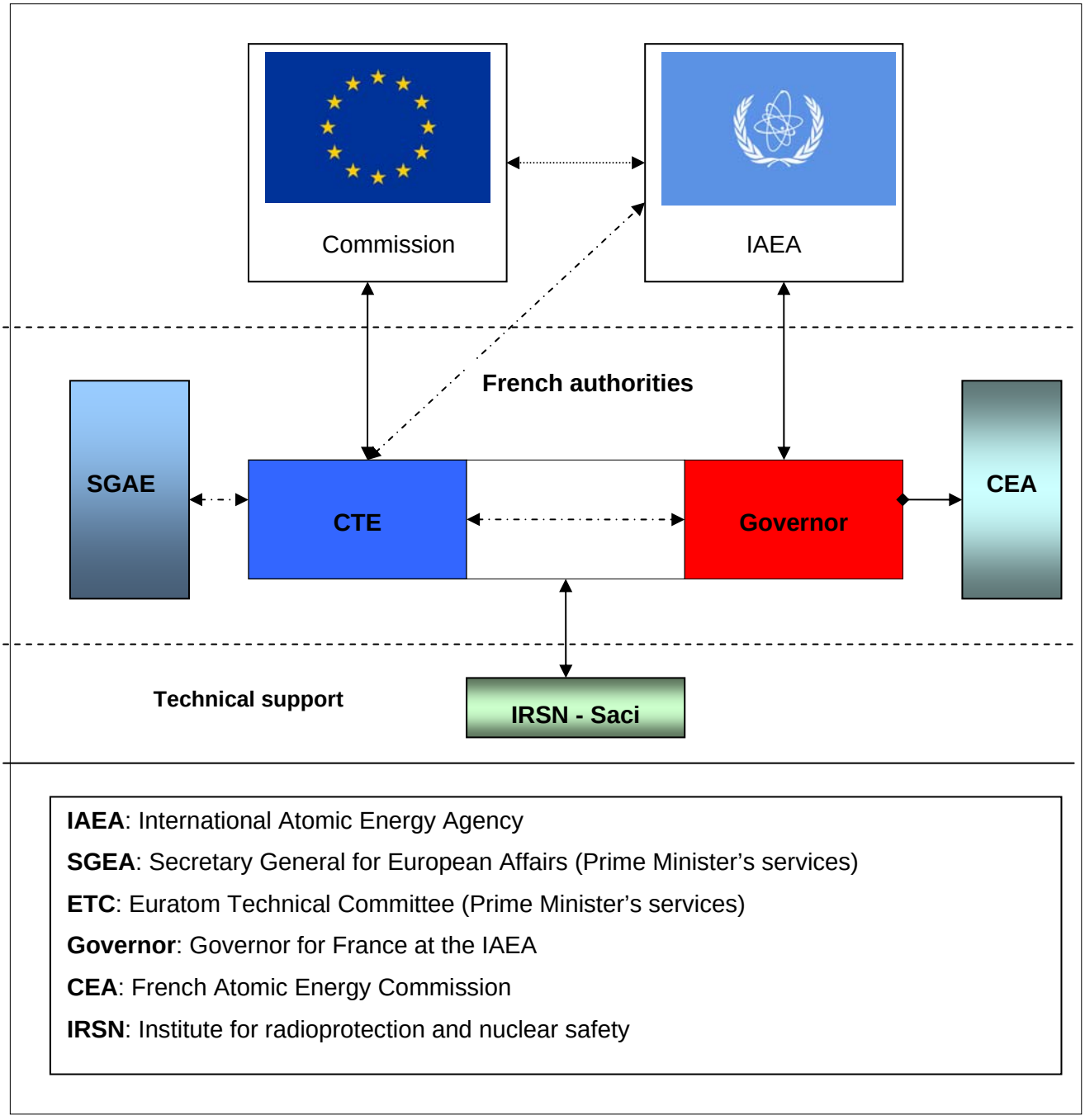
It should also be added that France regularly provides the IAEA with information in the frame of its additional protocol.

Thus with respect to the 2005 yearly declaration (sent to the IAEA in May, 2006), France counts 19 declaring entities for a total of 363 entries. The quarterly declarations correspond to a total of 127 entries.

## 2. The French organisation for monitoring international controls: a well coordinated set of varied participants

The enforcement of international controls in France is the subject of special monitoring in view of the large number of installations inspected, the materials held there, the existence of nuclear materials for (defence) national requirements, and the obligations undertaken by France.

The following diagram sums up the French organisation for the enforcement of international controls in France.



## 3. Perception of implementation of safeguards in France: a communication deficit

- **Perception by the national authorities**

Because of the number and diversity of installations, there is a high volume of safeguards implemented in France making it one of the most heavily safeguarded countries in the world.

This requires suitable national accompaniment, which is provided by the CTE and IRSN-Saci.

International safeguards conducted in France by the international bodies have overall relevance and rely on skilled staff. Improvements are of course possible, especially as regards limiting the constraints created for the operators. Improvement of interactions between the international inspection bodies must also continue.

- **Public perception in France**

The small world of international controls is little known to the general public. Even if certain aspects concerning international controls are brought up in public debates or in the media when there is talk of international proliferation crisis, it nonetheless remains a topic that has its main place in front of a "specialist" audience. Communication operations aimed at better presentation of the system of controls and their relevance are still necessary at national level.

- **Perception outside France**

International controls are on the whole well known to our European partners at least as far as the Euratom controls are concerned where these apply similarly in the European Union.

Outside the European Union, it turns out that the main concepts of the controls are known to partners with whom there are bilateral agreements (Euratom/third-State or France/third-State agreements). These controls are also known to certain industrial companies outside French borders, either because of a contractual impetus or because they belong to international groups.

Communication efforts remain necessary for better knowledge of the enforcement of international controls and the structures able to monitor this enforcement. These efforts must come from national level as well as from the international inspection bodies.

## 4. The French support programme (FRESPAS): 25 years of support to the Agency

On the international scale, France also supports IAEA in the area of safeguards through a support programme.

This support programme was officially created in 1982, just after France/Euratom/IAEA safeguards agreement came into effect.

In 2007, the FRESPAS celebrates 25 years existence and collaboration with the IAEA.

This programme is especially diversified in as far as it covers

- equipment
- analyses
- expertise
- satellite imagery
- training
- universalisation of safeguards

FRESPAS actions were assessed at 1.2 million euros in 2006.

They involve a lot of French nuclear actors: CEA, AREVA, IRSN, INSTN, LCPC.

Programme coordination is by CEA's Department of International Relations. The CEA is furthermore the biggest contributor to FRESPAS.

It is clear that French skill and expertise in the nuclear field, especially its mastery of the whole nuclear fuel cycle (including reprocessing or Mox manufacture), is of great interest to the IAEA.

- **Overview of the FRESPAS tasks**

The French support programme comprises between 20 and 25 active tasks in a given year.

As an example, the following tasks are implemented within the scope of the FRESPAS:

. Assessment of non-destructive geophysical methods. This means helping interpret the results from “georadar” type measuring equipment for detecting undeclared nuclear activity (underground installation or dissimulated cavities, etc.). The CEA Dept. of Military Applications and Dept. of International Relations are involved in this task as is the “Laboratoire central des ponts et chaussées (LCPC)”.

. France also provides on-demand expertise to the IAEA. France, for example, made a cost free expert available to the Agency assessing safeguards costs. This expert is from the Institute of Radioprotection and Nuclear Safety (IRSN).

. As part of the inspection programme for the Rokkashomura reprocessing plant in Japan, an AREVA expert has been made available as cost free expert to the Agency so it can reap the benefit of the experience acquired in France in running reprocessing plants. Help is now provided in the form of a programme of familiarization visits by IAEA staff to the La Hague processing plant and the Mox production plant (Melox). France also finances the participation of IAEA staff in a training session on the nuclear fuel cycle organised by the National Institute of Nuclear Sciences and Techniques (INSTN).

. A laboratory operated by the CEA's Department of Military Applications is vetted by the Agency and undertakes analysis of samples taken by it. France also participates in comparison circuits for samples of Plutonium and Uranium through CETAMA, which answers to the CEA's Department of Nuclear Energy (DEN).

. France also supports Agency actions aiming at making safeguards universal. It participates in financing certain seminars organised by the Agency for promoting signings and the implementation of revised small quantities protocols or additional protocols.

- **Results of 25 years supporting the Agency**

The collaboration between France and the Agency is highly satisfactory as much for the human aspects in managing the support programme as for the technical aspects. This collaboration is even more fruitful in as much as France doesn't just help finance actions but also puts expertise and skills at the service of the objectives set by the Agency.

The French support programme has always enjoyed political support in France.

It has nevertheless evolved over the years. It was initially dedicated to the development of equipment, until the last few years when it has diversified greatly. Moreover, it brings together a whole host of French partners.

The programme has however essentially remained a French one through the years, even if the prospect of collaboration with other support programmes is starting to see the light of day.

Its richness resides in the quality of French expertise and good knowledge of the whole nuclear fuel cycle.

These 25 years have of course not been entirely synonymous with success. Certain tasks have proved impossible to achieve. But the great majority of them have produced convincing results.

- **Difficulties encountered**

Certain FRESPAS tasks in the past have created constraints for operators when installations, experts, or equipment have been made available to the Agency.

The implementation of the FRESPAS also sometimes comes up against sensitivity issues with certain information and technologies, which does not always make it easy to give a positive reply to IAEA requests and which calls for a suitable framework for managing the confidentiality of information.

The financing question is also important for the FRESPAS. French financing has changed since the programme was set up. It was initially provided by the ministries whereas now it is mainly provided by public bodies like CEA and IRSN or in the case of specific tasks by companies like AREVA.

Lastly, IAEA communication on the worth of the support provided by the programmes needs reinforcing in order to get the French experts investing in the Agency's works and give new French partners an incentive to join the FRESPAS while ensuring the ongoing nature of financial contributions.

- **FRESPAS prospects**

France intends to continue its support to IAEA as long as IAEA's needs are real and have not decreased.

Nonetheless, the support programmes must not be considered more than a complement to actions mainly financed by the Agency's regular budget.

As regards short and medium term prospects, France wants to get a second analysis laboratory endorsed in the network of analytical laboratories at the service of the Agency.

France is also involved in deliberations on the introduction of new technologies, which represents one of the technical and legal challenges in the coming years as regards modernising and further strengthening the Agency's capabilities in the detection of undeclared nuclear activity.

The FRESPAS could in the future also be more open to collaboration with other support programmes, for example that of the European Union.

New French partners will also be encouraged to join the FRESPAS.

# Finnish Support Program to the IAEA Safeguards from 1988 - 2006 - History, Importance and Future Perspectives

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***Abstract:***

*Finnish Support Program to the IAEA Safeguards (FINSP) was established in 1988 and it is funded by the Ministry for Foreign Affairs. The ongoing support has been a consistent commitment by Finland to work for non-proliferation. In Finland it is understood the IAEA appreciates R&D and training support from the Member States in order to accomplish its functions efficiently and effectively. The annual program budget for 2006 is 190,000€.*

*During the past years the strong areas of FINSP have been development of new verification methods and safeguards training. In this paper four highlights of FINSP task areas are presented in more depth 1) Development of BWR Spent Fuel Attribute Tester, 2) Development of Tomographic verifier 3) Additional Protocol Complementary Access training for the Agency inspectors and 4) Development of Final Disposal Safeguards*

*Lesson learned from the previous experience is that sufficient understanding of IAEA's needs is a key to success. In an ideal situation the IAEA clearly identifies how it best benefits from the SP tasks and experience and expertise FINSP has to offer. FINSP has light administration structure but on the other hand it has wide network of experts at its disposal. This makes FINSP flexible and fast to respond. Several cases have shown that the success of the implementation of FINSP highly depends on the common understanding based on in-depth identification of momentous IAEA needs.*

*FINSP is also an important cooperation channel between Finland and the Agency. Mutual contacts have greatly helped Finnish SSAC and the IAEA also in the practical implementation of the INFCIRC 153, 193 and 540 type agreements.*

*In the future FINSP continues to concentrate in its traditionally strong areas. Development of new verification methods and inspection tools will remain in focus together with support to IAEA inspector training.*

**Keywords:** Support Program to the IAEA Safeguards; Safeguards R&D; Safeguards Training, NDA

## 1. Introduction

Finnish Support Program to the IAEA Safeguards (FINSP) was established in 1988 and it is funded by the Ministry for Foreign Affairs. The ongoing support has been a consistent commitment by Finland to work for non-proliferation. In Finland it is understood the IAEA appreciates R&D and training support from the Member States in order to accomplish its functions efficiently and effectively. The annual program budget for 2007 is 190,000€.

During the past years the strong areas of FINSP have been development of new verification methods and safeguards training. This paper gives four task area highlights where FINSP has been particularly active in the past years: 1) Development of BWR Spent Fuel Attribute Tester, 2) Development of Tomographic verifier 3) Additional Protocol Complementary Access training for the Agency inspectors and 4) Development of Final Disposal Safeguards

## 2. Case areas

### 2.1 BWR Spent Fuel Attribute Tester

In the 1980's it became evident that the IAEA should start to employ new verification methods in wet spent fuel storages. Cerenkov Viewing Devices (CDV's) commonly used in spent fuel pools was not always the feasible method. In some cases the poor water quality prevented the use of CVD. It was also understood that CVD is indirect method, because it did not detect the actual attribute of the spent fuel but visible photons created in the water by beta particles.

The clear benefit of CVD was, however, its acceptability by the operators. No fuel movements are needed and nothing is introduced into the pool. It was in the beginning clear that no such NDA methods based on radiation measurement could be invented. In order to detect radiation fingerprints the spent fuel the measuring device should be lowered close enough to the spent fuel assembly so detectable signal could be acquired. Spent Fuel Attribute (SFAT) method was the most promising to solve major agency problems, since the principle was simple and no fuel movements were needed.

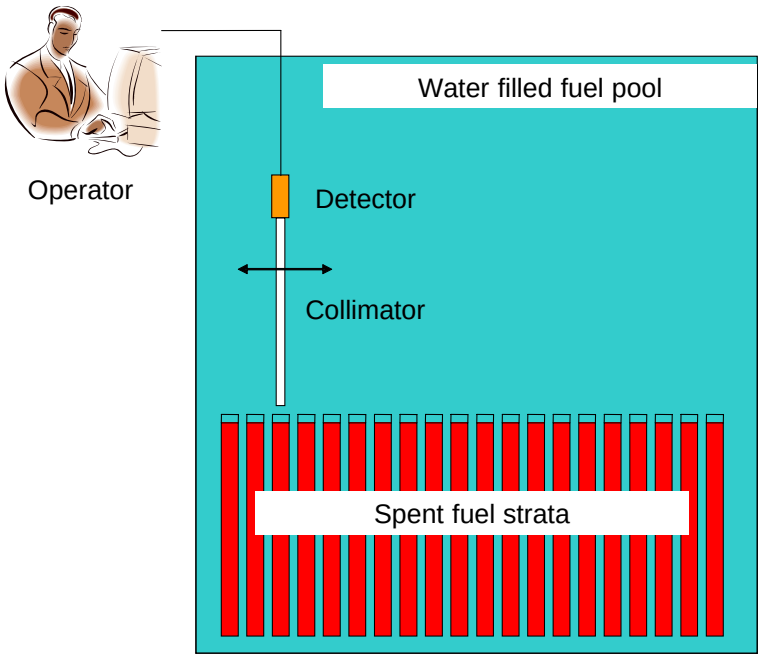


**Figure 1** Principle of SFAT method. The air filled collimator is lowered down close to the assemblies in the storage pool. The distance should be minimal, but touching the assemblies is not desirable. The gamma radiation is detected by the detector at the top of the collimator.

The FINSP accepted the Agency Task JNT563, to develop universal SFAT device for the IAEA use in BWR type facilities. Lots of these reactors were and still are operational in many European States.. To ease the operation and licensing the device was decided to be operated by the fuel transfer machine and the appearance of the device was designed to be similar to a spent fuel assembly.

At that time Sodium Iodine (NaI) scintillator detectors were only feasible type of spectrometer in this kind of application.

A Finnish private company, Plustech, participated the project alongside with VTT, Technical Research Centre of Finland. Also USSP participated into the task. Los Alamos National Laboratory performed very useful Monte Carlo calculations needed to optimize the device geometry and the size of the detector. Power company TVO gave also comments and provided the test site for the prototype.

Under the task JNT563 task a commercial prototype of BWR SFAT was constructed. The prototype was delivered to TVO KPA-STORE where was successfully tested and the final report by Tarvainen et al [1] was submitted to the Agency.

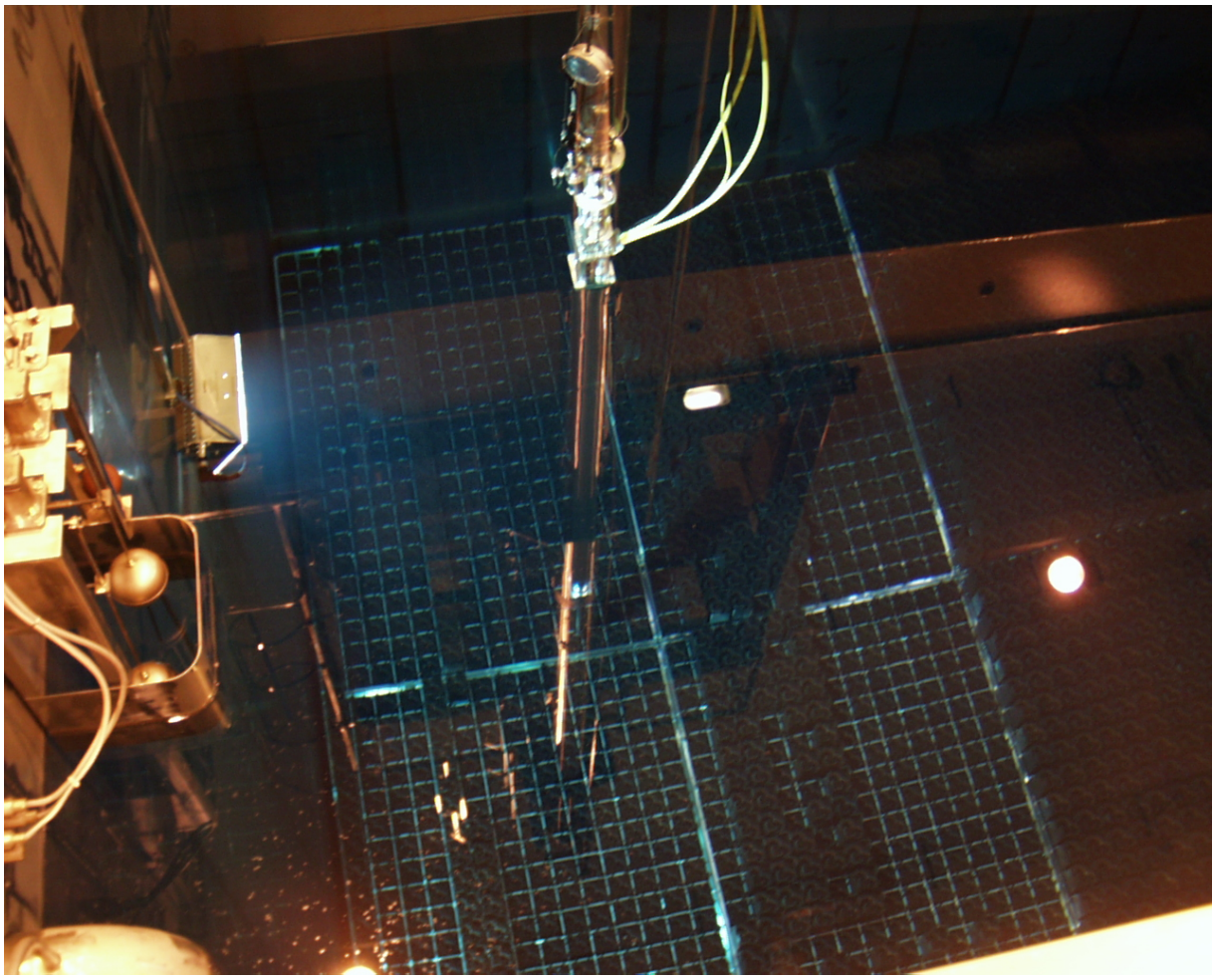

**Figure 2:** BWR-SFAT commercial prototype being operated in Olkiluoto Spent fuel storage pool.

The prototype device was designed by a Finnish private Company Plustech, later joined into John Deere inc. The design was rather complicated due to the fact that Olkiluoto Spent fuel store, where the device prototype was to be tested had two types of Spent Fuel Assemblies (SFA's). In the early days of operation TVO recirculated fuel channels from one SFA to another. Therefore some SFA's in the pool have a fuel channel and some not. The SFA's having a fuel channel are about 110 mm taller than those without a channel. Since the successful verification provided that the distance at the end of collimator to the handle of the SFA's is equal or less than 100 mm there was clearly a risk that the SFAT will touch the SFA's with a channel during the operation. To minimise this risk the operator requested that the system should have a distance gauge to accurately determine the distance to the SFA. Also the SFAT, attached to the fuel transfer machine should be operated at the constant level. The detector-collimator part of the device is vertically lifted and lowered with a stepping motor mounted inside the SFAT.

The requirements made the device rather bulky and electro-mechanically complex. However, the prototype was successfully produced, with accurate documentation. It has been used by STUK and the IAEA since 1993 at Olkiluoto in many campaigns. In 1998 the device was badly damaged during a campaign, its cable lead-ins started to leak and internals of SFAT became filled with water. However, it was repaired and serviced in 1999. The device is still successfully used in Finnish national verification programme. Actually it is at the moment at its strongest: in 2007 it was thoroughly serviced and the detector was upgraded to 1500 mm3 CdZnTl semiconductor which greatly enhanced its detection capability. Although the IAEA has not directly used the device it has served many times in the Agency training courses. FINSP is willing to provide continuously training opportunities for the IAEA staff and the BWR SFAT is one of the tools.

The Construction of BWR SFAT gave the Agency also important information and experience, which were useful when the IAEA designed its own lightweight SFAT.

The BWR SFAT shows that even rather complicated development project can be successfully carried out, if carefully planned and the device can be kept operational over a decade, if it is constantly used and serviced. The latter is not possible without sufficient documentation.

## 2. 2 Passive Emission Tomographic Verifier

At the moment measuring method development task JNT 1510, "Development of Passive Emission Tomographic Verifier". is a priority. The method was developed already in the beginning of 1990's by some research groups. Several tests were performed earlier under various SP tasks (Lévai et al. [2]). The development of detector technology has made it possible to construct a detector array needed for measurement of even large diameter PWR fuel assemblies.

The IAEA has a need for the device, since it needs ultimate verification tool at its disposal. With the tomographic verifier it will be possible to perform pin-level verification of any present type fuel assembly. Also other types of fuel items can be verified. The device produces cross sectional activity map of the verified item. Also gamma spectrum will be acquired. Because of the high price of such an unit it is expected the IAEA will not procure many of these devices, but it will have an important role in resolving special cases.

JNT 1510 started in 2005, when the IAEA made the decision to finance the detector arrays. At the moment the detector arrays are being manufactured. However the contracted company has got some delays, which may postpone the delivery of the verifier. According to the original plan the device could have been tested at the end of 2007 (Honkamaa et al. [3]).

The project is a good example of good cooperation between MSSP's. The need of the IAEA is clearly identified and they are addressed during the design of the device. The construction of the prototype requires high level of scientific, software development and mechanical skills, and also contacts to the suitable facilities which can offer appropriate site and materials for testing purposes. No single Member State has all these capabilities and facilities at its disposal, so joint effort is the only way to make successful progress.

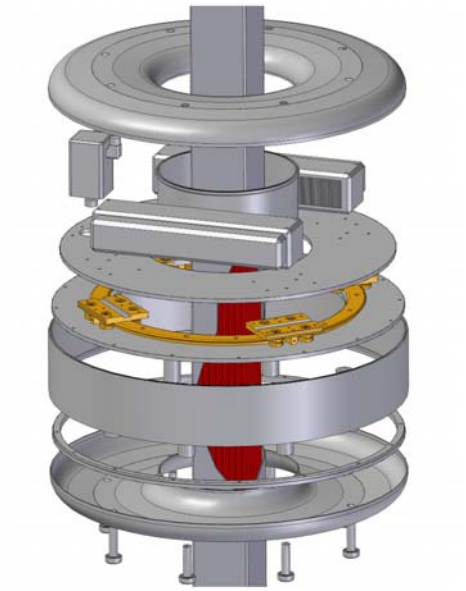

**Figure 3:** Schematic picture of the Passive Gamma Emission Tomographic Verifier. The verifier will be toroidal in shape and the item being verified is placed inside the torus. Two detector arrays, each of which having 104 CdZnTl detectors, rotate around the torus generating data for tomographic image.

## 2. 3 Additional Protocol Training for the IAEA inspectors

After the negotiation of Additional Protocol it became evident that the IAEA should train its inspectors for the new regime. The IAEA has now a new tool at its disposal, Complementary Access, Since this is to be widely applied in Member States signed and ratified the Additional Protocol the Complementary Access should be performed by the inspectors, which previously were conducting traditional safeguards inspections only.

CA requires completely new thinking. The mandate of the inspector and the IAEA must be clearly understood. New methods and tools should be learned. The emphasis is on soft skills.

The need of specific training course for the IAEA inspectors was quickly made note of. However, in the beginning it was not clear even for the IAEA how this training should be conducted. FINSP made a proposal in 2003-2004 to the IAEA about organising field part of a training course in Finnish facilities. IAEA studied the proposal and found it useful. If sent SP1 to the FINSP, which was accepted as FIN B 1422 “Workshop on Additional Protocol Activities”. IAEA drafted together with FINSP a draft schedule for the course, which was tested in a preliminary course in 2004. The participants in this course constituted the trainers in the first actual course. Finland provided both a research centre and NPP a for training. The cooperation of Finnish Technical Research Centre and Fortum Power and Heat is acknowledged. By now altogether 5 courses have been organised in Finland. IAEA has requested FINSP to provide more courses, the next one is planned to take place in November 2007.

The ongoing need of IAEA inspector training calls for more training products. A training video is being produced, which describes and the most intensive learning experiences obtained during the courses. The video is expected to be ready in the fall 2007.

**Figure 4:** Complementary Access Training Video for the IAEA being filmed in STUK, May 2007. Cameraman Ville Mäntymaa from Publishing House Koivuniemi is recording the actions of the main film stars Elina Martikka (STUK) and Zoran Vukadin (IAEA).

### 2. 4 Development of final disposal safeguards.

Geological repositories are a new challenge for the traditional IAEA safeguards. In the disposal process nuclear materials are encapsulated in such manner that the item counting and identification can be focused only on the canisters containing the declared materials, and finally when emplaced in the geological medium, no re-verification is adequate. Therefore, new safeguards approaches have to be developed. This was initiated by the DG in 1989 and a corresponding task was formulated a few years later.

FINSP has been active in participating in the SAGOR I and SAGOR II phases of the joint task Safeguards for the Final Disposal of Spent Fuel in Geological Repositories during 1994 - 2006. The SAGOR phase I produced an extensive generic approach for the application of safeguards to geological repositories. The Finnish Support Programme was responsible for the subtask to develop verification methods (Tarvainen et al. [4]). The SAGOR phase II concentrated on the site-specific application of the generic approach providing also advice about new technology. During the Phase II the site-specific issues of the Finnish repository site at Olkiluoto were discussed and the site itself was visited twice. The methods applied in the Finnish site investigations were demonstrated (Okko [5]), analysed and reported for their applicability in safeguards (Okko & Rautjärvi [6]).

FINSP has also created a preliminary safeguards concept for final encapsulation plant in Olkiluoto (Rautjarvi et al. [7]). The concept gives a basis for further development, which helps, *inter alia*, to identify technical research needs and the roles of different organisations.

The SAGOR programme is presently followed by the ASTOR-task, which gives information about the application of safeguards to repositories. The FIN SP will provide the ASTOR task with updated information about the progress in the Finnish repository project. The new methods and approaches to be applied in national and international safeguards system are described by Okko & Rautjärvi [8].

The ASTOR programme is scheduled for a 10-years period which also reflects the long-term aspects in the development. During the ASTOR phase the Finnish project is supposed to advance from a pre-nuclear research phase to a licensed pre-operational phase. This provides the Agency with a unique opportunity to apply the generic safeguards approach from the SAGOR phase in the implementation of safeguards to geological repositories.

## 3. Future of the FINSP

Ministry of Foreign Affairs as a funding organisation has acknowledged the importance of FINSP and the program will most likely continue also in the future. The rapid expansion of the scope is not foreseen, but the training tasks and development of measuring methods will continue.

From the point of scientific and intellectual capacity in Finland the FINSP could have much more to offer to the IAEA. The Finnish research organisations and private companies have lots of relevant expertise to offer for the IAEA. The relatively small size of Finland makes it easier to establish networks within the country. FINSP will therefore service as an important contact channel between the IAEA and Finnish companies and organisations. Only the budget limitation prevents the expansion. At the moment the projects must be carefully prioritised.

Mutual contacts have greatly helped Finnish SSAC and the IAEA also in the practical implementation of the INFCIRC 153, 193 and 540 type agreements.

Lesson learned from the previous experience is that sufficient understanding of IAEA's needs is a key to success. In an ideal situation the IAEA clearly identifies how it best benefits from the SP tasks and experience and expertise FINSP has to offer.

## 4. Conclusion

In the future FINSP continues to concentrate in its traditionally strong areas. Development of new verification methods and inspection tools will remain in focus together with support to IAEA inspector training. New verification tools are actively search for, which can be useful for the IAEA:

With the discussions with the Agency Support Programme administration it has been understood that also small MSSP's has great strategic value for the Agency. The credibility of SP concept relies on the fact that there is a variety of different states participating the program. Small MSSP's usually have light administration, which makes them fast to react emerging needs of the IAEA. Our experience is that the IAEA's Support Program Administration fulfils its expectations efficiently and in good cooperation.

We want encourage old MSSP's to contribute also in the future and new MSSP's to join the program. The costs of the program are small compared to the benefits for both the IAEA and contributing MSSP.

## 5. Acknowledgements

The authors would like to acknowledge the earlier FINSP coordinators: Matti Tarvainen, Juha Rautjärvi and Erja Kainulainen for their contributions. Vera Firbasova from the IAEA SPA provided some valuable information. The last but not least the support of Foreign Ministry of Finland and comments from Councellor Tarja Pesämaa is gratefully acknowledged.

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# Swedish support programme 1992-2006: Advanced Systems for Safeguards Information Treatment in Ukraine

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***Abstract:***

*The work of developing systems for Safeguards Information Treatment in Ukraine has been going on since 1992. The work has been a part of the Swedish Support Programme to Ukraine.*

*Under the Swedish support programme, several programs have been created between 1992 and 1997. These programs were written for operating system DOS.*

*Due to the quick development of sophisticated software tools and the improved knowledge of safeguards in the NIS, a new series of software have been developed. They where called the 3rd generation Advanced Integrated accountancy systems. These systems have been delivered under the Swedish support programme, between 1998 and 1999.*

*During the period of 2000 to 2005 these software have been completely rebuilt using fully object oriented high tech tools. These new program series is called STAR. The STAR software is the result of 20 years of evolution and development concerning high tech data treatment. The new software for the NPP's includes special functions like advanced fuel calculations and maps. For Zaporozhye NPP a completely new program for the Spent Fuel Storage was created during 2005.*

*The STAR software for the state authority was delivered in year 2000. This software has been upgraded during 2003 with new functions, for example handling of LOF's. During 2006 further developing of the system with advanced QC functions and a new Management System for extended control and analysis were included.*

**Keywords:** SSAC; NMAS; support; software

## 1. Introduction

In 1992 the Swedish Nuclear Power Inspectorate invited participants from the NIS to attend a Safeguards Seminar in Stockholm. This was the beginning of the Swedish support programme and of the cooperation between Sweden and Ukraine.

## 2. The support programme

The work started with the development of principles and regulations for a manual SSAC system. After completing this stage the work continued with making the first software for Safeguards Information Treatment at the state office. This step also included the creation of a layout for advanced data communication.

The principles for the communication system were to receive all inventory change data and all inventory data in ASCII text format files. The structures of the files were defined in cooperation with the Ukrainian State Office. Using predefined file structures for reporting gives the opportunity to send a

large amount of data from the facilities to the state office and between the facilities within the country. The files can be sent on diskettes or by E-mail.

The main task for the State Office program system was to collect all relevant information from all facilities in Ukraine and compile the data into the form required by IAEA. All reporting from Ukraine is checked and sent from the State Office to the IAEA.

After the development of the first program for the State Office it was necessary to start the development of software for use at the facilities. The former Soviet systems at the facilities only handled the inventory and did not include a General Ledger for inventory changes. During 1993 to 1995 basic General Ledger handlers was delivered within the Swedish support programme to NPP's in Khmelnitsky, South Ukraine, Chernobyl, Zaporozhye, the research reactors in Kiev and Sevastopol and to Kharkov Institute of Physics and Technology. During this period advanced SSAC systems where also delivered to the state offices and facilities in Kazakhstan and Lithuania.

In connection with the development of the software an extensive training program were conducted. The training was held for personnel at the state office and for operators at the facilities. The training covered all topics from initial safeguards courses to advanced SSAC.

Based on the Swedish assistance and close cooperation Ukraine had possibility to start the functioning of SSAC since 1994 and fulfilled the obligations according to Safeguards Agreement ( type INFCIRC/153) which entered into force in the beginning 1995.

The 2$^{nd}$ generation Advanced Integrated accountancy system was first created for Khmelnitsky NPP and delivered in 1996. The system was built to handle all aspects of information treatment at the facility with a minimum of data entrance to minimize errors. This software included a General Ledger, an Inventory and a Fuel history. The system also included sophisticated QC methods. This system was written in DOS with a graphical interface. Later this year a system for South Ukraine was also delivered. Zaporozhye NPP got the new software in 1997.

During these years the cooperation also aimed on fulfilling the needs of hardware necessary for running the accountancy systems in use.

Due to the quick development of sophisticated software tools and the improved knowledge of safeguards in Ukraine a new series of software were planned.

During the period of 1998 to 2000 the new program series was created. This new 3$^{rd}$ generation Advanced Integrated accountancy systems is using the latest fully object oriented high tech database tools. The new advanced program series is called STAR. The STAR software is the result of nearly 20 years of evolution and development concerning sophisticated high tech data treatment. The new STAR program gives extraordinary capabilities for treatment and analysis of information. The STAR software fully uses the advanced capabilities of the modern Windows environment.

The STAR software was first introduced at the Swedish NPPs Barsebäck and Oskarshamn.
The new STAR program for the State office in Ukraine was created in close cooperation with the state authority and was delivered in year 2000 together with similar programs for the State offices in Lithuania and Latvia. The first STAR software was presented at a poster session at Esarda 2001 in Brugge.

The new STAR programs for facilities were delivered to South Ukraine NPP in 2001, Khmelnitsky NPP in 2002, Zaporozhye NPP in 2003 and to Rowno NPP in 2004. These programs have been updated regularly, based on the results of the close cooperation with the operators.

During 2005 the software for the State Office has been updated with an enhanced Quality Control system, functions for advanced automatic error correction and an automatic logbook. New special software for the Spent Fuel Storage in Zaporozhye has also been developed and delivered in 2005.

Extensive training has been held in accordance with the delivery of each new software system.

During 2006 a new Management Software for the State Office was developed. This software makes it possible to increase the usage of the accountancy data for other purposes than just reporting, and

connect it to other sources of data.

The latest project covers the development of a complete education package for the university in Sevastopol on the STAR NPP. This will give students in the nuclear field a possibility to learn safeguards and how to use an advanced accountancy system.

## 3. Conclusions

The cooperation between Sweden and Ukraine has been going on for more than 14 years. Sweden was the first country to get involved in the field of NMAC in Ukraine.

Maybe the most important part of the cooperation has been the continuous work with maintenance of delivered software together with technical and safeguards support combined with a close dialogue with the State Office and the facilities. This is now days called “sustainability”.

The main principles for our successful cooperation:

- Best possible NMAC customized software for the different tasks
- Extensive training
- Providing continuous maintenance and support of delivered software
- Ongoing dialogue of advanced NMAC theory and applications
- Creating a person to person relationship with the people using the software, to facilitate communication and ability to provide fast assistance when help is needed.
- Possibilities for students to learn NMAC to secure the quality of the SSAC in the future.

## 4. Additional information about the STAR software

The STAR programs are developed for use on PC´s. The programs are written in Visual Objects which is a 32 bit fully object oriented database language for Windows 95/98/NT/2000 and XP.

The software is designed to take into account the account the special needs of the national accountancy system. The program performs the reporting to IAEA in accordance to the country's specific agreement.

The programs have been designed to be a powerful working tool for the operators. The software is designed to be extremely easy to handle and at the same time give the operator the maximal power of searches, calculations and reports that a modern advanced system can produce.

The system is designed to read and generate the specific advanced communication files that are used in Ukraine. This minimizes the amount of manual input which is one of the main sources for errors in the reporting.

All programs are designed for a multi-user environment in a network with different access levels.

### 4.1. General features:

- Module Built to be able to easy add new functions
- Menu driven System
- Minimum data entering requirements
- Data bases are automatically updated
- Advanced QC System
- Powerful Filter Functions (Enhanced SQL type)
- Possibility to export/import data from other languages
- Data easy to access for other applications
- Automatic creation of necessary documents
- Custom Made Reports

- Designed for Advanced electronically communication

## 4.2. STAR for SSAC including the Management system

STAR for the state office is specially designed for the specific Ukrainian national system.

The program receives data from the different facilities in the country electronically by special files. These files update the General Ledger and the Inventory automatically. All data are checked by the built-in Quality Control system before they are loaded into the databases.

The program is capable of advanced searches and calculations of the inventory and the flow in all facilities. This enables the personnel at the State Office to have an overview of the nuclear material in the entire country.

The system also handles all LOF's (Location outside facilities) in Ukraine. This covers the reporting requirements for all institutions handling small amount of nuclear material.

All relevant reports to the IAEA are created automatically by the program and can be sent electronically to the agency.

With the Management system is it possible to make "global" searches and calculations, i.e. to be able to calculate the total amount and/or making advanced searches of material in the entire country or selected parts. This is an essential tool to have control over the entire SSAC and is helpful in decision making.

The Management system automatically keeps track of all incoming and outgoing accountancy reports. All reports are instantly available for viewing. This enhances the control over the required data and report flow.

## 4.3. STAR for Facilities

The systems for the facilities are specially designed for the different types of reactors and storages. These systems were initially designed to maintain the "knowledge" from the old Soviet accountancy system, without need of reproducing the papers and principles of that system.

The STAR program includes a General Ledger, an Inventory, a Fuel History, Material Balance Reports and a Logbook. All data bases are updated automatically based on the actions of the operator.

For Inventory Changes the receipt are entered from a file or manually, unless it is a receipt from another MBA at the same facility then the receipt is entered automatically.

To make a shipment it is only necessary to select the fuel to ship and the program will generate all relevant records and update all data bases.

The program is capable of calculating the new post-irradiated content of the spent fuel based on the actual burnup, fuel type and enrichment. All relevant accountancy is produced automatically.

Fuel movements are entered by giving the new coordinates for the fuel assembly manually or from a file. The program will check that the new coordinate exists and is empty. Then all relevant data bases are automatically updated and Inventory Change records are generated in the case of movement of Spend fuel into the core. The program also updates the relevant core and pond maps.

The software is also capable of making rebatching of fuel assemblies when loaded into fuel casks. All data is automatically updated and relevant reports are generated.

Advanced searches and calculations of the inventory, General Ledger and the fuel history are possible. This gives the operator a powerful tool for evaluation of the inventory.

All activities are automatically logged in a logbook, so the operator can keep track of previous

activities.

The program generates all necessary reports, including shipping documents and maps for core and ponds. All information required by IAEA and the state authority can easily be produced.

Zaporozhye NPP also has special software for the spent fuel storage. This program is similar to the program for the reactors but it focus on the accountancy for the spend fuel containers. The software also contains a graphical interface for the movement of fuel containers.

All relevant information for the fuel casks are kept in a General Ledger and the inventory. Detailed information about the fuel assemblies inside the fuel casks is kept in a special data base.

### 4.4 Delivered STAR programs

The following STAR programs have been delivered to Ukraine:

- Khmelnitsky NPP
- Rowno NPP
- South Ukraine NPP
- Zaporozhye NPP and Dry Storage
- SSAC system for SNRCU
- Management system for SNRCU

# Session 5

# DA

# The Safeguards Analytical Laboratory and the Future of Nuclear Materials Analysis for the IAEA

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**Abstract:**

*In order to fulfil an analytical obligation for between 1500 and 5000 nuclear material samples per year, in the early-1970s the IAEA established a two-prong system consisting of a network of analytical laboratories (NWAL) and a dedicated safeguards analytical laboratory (SAL). For reasons of economy and transport expediency, an Austrian-based SAL was incorporated into the NWAL to provide routine analytical service, while reserving part of its work to support a quality control program and to investigate particular problems upon request from IAEA Safeguards staff. Since its inception in 1975, the SAL has carried out nearly all of the nuclear material analytical requests for the IAEA with notable consistency, quality and value. This arrangement has suited the IAEA well for over 30 years, however as the infrastructure of the existing SAL ages, questions regarding the future of this arrangement have been raised. In November 2006, a workshop of technical and policy experts from the IAEA and concerned member states was convened at the Agency's Laboratories in Seibersdorf, Austria to address the requirements of the IAEA for safeguards-related analytical services. The importance of a dedicated laboratory for IAEA safeguards analytical work was re-emphasized. However, attendees also agreed that greater reliance on qualified NWAL, and perhaps non-NWAL facilities, for nuclear materials analysis should be studied. In this context, the IAEA will seek to re-establish the nuclear network of analytical laboratories, which has fallen dormant over the last decade. In parallel, the Agency will investigate the feasibility of sample shipment logistics (e.g. timeliness) and cost. A subset of labs will then receive a set of characterized nuclear material samples for analysis, against which quality, timeliness and value to the IAEA will be judged. Furthermore, the Agency will begin to objectively evaluate renovation or reconstruction options for the SAL.*

**Keywords:** nuclear material sampling, safeguards, analysis, network laboratories

## 1. Introduction

The IAEA Safeguards Analytical Laboratory (SAL) has as its principal mission to provide the IAEA Department of Safeguards with accurate, precise and timely analytical results from nuclear and environmental samples collected by Safeguards inspectors worldwide. SAL has fulfilled its role as service provider to Safeguards since the 1970s, in cooperation with a network of analytical laboratories worldwide. In the field of nuclear material sample analysis, SAL strives to improve methods that are cost- and time-effective and deliver results of high quality. In the analysis of environmental samples, SAL works to improve both throughput and timeliness and to perform measurements of samples with very low amounts of U, Pu and other elements with high precision and accuracy by either bulk or particle analysis.

In addition to its principal analytical mission, SAL is tasked to provide materials in support of inspection activities (e.g. sample bottles, environmental swipe kits, and elemental/isotopic standards), and coordinates much of the collaboration within the Network of Analytical Laboratories for analytical support in both nuclear and environmental sample analysis. Figure 1 provides a historic overview of samples received at SAL. In the year 2006, 756 nuclear material samples have been received, 760 samples have been analysed and 1664 verification results have been generated for these samples. The figure also shows the increase and stabilisation in environmental samples received, 492 in 2006.

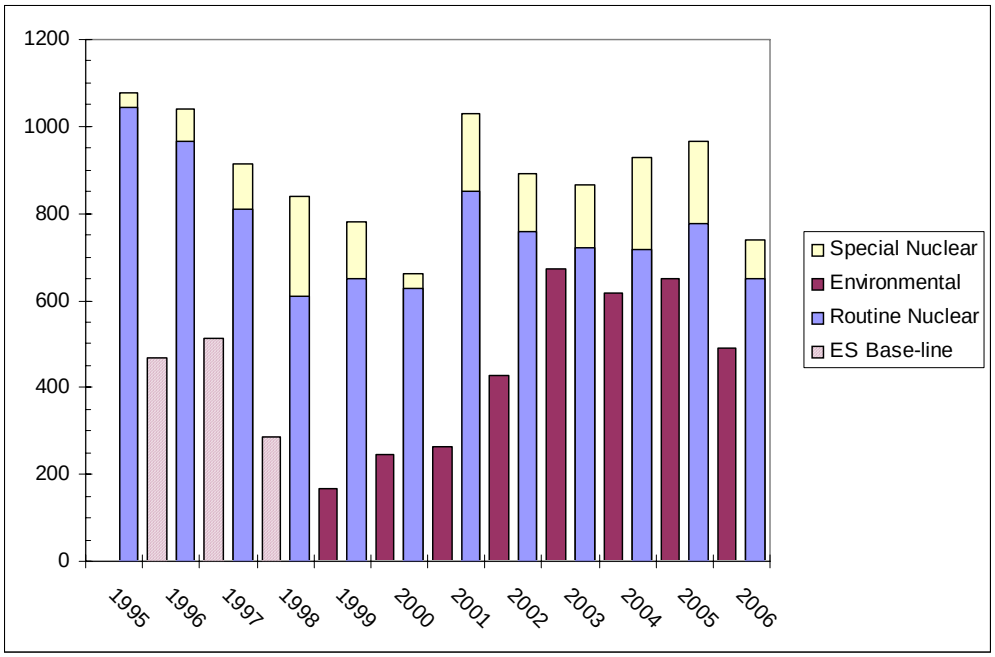


**Figure 1**: Nuclear Material and Environmental Samples received at SAL from 1995 to 2006

By far the largest fraction of nuclear samples is analysed for Uranium (450~500), much smaller fractions are analysed for Plutonium, handled in the Input Lab (input or high active liquid waste solutions), or treated as special samples. As of now, only Heavy Water samples are analysed in the network.

## 2. SAL and the Network

### 2.1. Status Quo

As already indicated, almost all of the nuclear material samples get analysed at SAL. Though the network of analytical laboratories was originally designed to share the load of nuclear sample analysis, it has evolved to become a network for environmental sample analysis predominantly. After receipt and initial screening of environmental samples, approximately 80% thereof are analysed in the network. Figure 2 shows a comparison of the load distribution between SAL and the network.

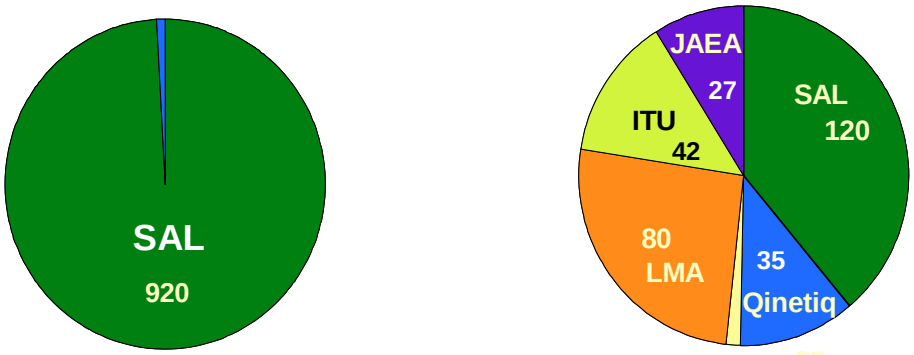


**Figure 2**: Distribution of nuclear material samples in the network (left) and distribution of environmental SIMS samples in the network (right) – data taken from 2004.
Whereas the distribution of SIMS-samples throughout the network shows a balanced situation, the state of affairs regarding nuclear samples does not share this characteristic

### 2.2. Safeguards Needs

The Department of Safeguards has analysed its analytical mid-term to long-term needs, based on foreseeable developments and facility specific information. It is their understanding that – apart from some fluctuations – the average amount of nuclear and environmental samples should remain stable at current levels. This already considers the dis-continuation of MOX production by Belgonucleaire and the shut-down of the Tokai Reprocessing Plant in the long term, but also some more unclear assignments probably waiting in line for Safeguards, such as North Korea or India. In the opinion of Safeguards, it would further be prudent to accommodate additional peak load capacity for unforeseen developments. It has been re-emphasised that destructive analysis – accurate, timely and confidential – is an indispensable element of deriving Safeguards conclusions. The best service in fulfilling this need is seen as a strong and independent IAEA facility, supported by a strong network. Since currently this situation may only be found for the analysis of environmental sample analysis, the situation for nuclear material samples needs addressing.

### 2.3. Desirable Situation

For nuclear material analysis, the current situation is far from perfect. Two issues need to be considered:

- The nuclear part of the network has fallen almost dormant over the last decade. Current capability and capacity of the network needs to be revisited and possibly expanded.
- SAL itself is suffering from ageing infrastructure resulting in concerns as to the sustainability of operations in the mid-term future. This will be addressed in the next section.

Further support from the network will be required for both nuclear material and environmental sample analysis. While for environmental samples the network is working well and only enhancements in throughput and timeliness are desirable, more needs to be accomplished for the nuclear material network. The needs may be characterised as follows:

- Provide analytical capacity for all sample types in times of high demand, thus improving the average response time for reporting the results of inspection sample analysis.
- Serve as external quality control of SAL's analysis results by measuring replicate (parallel) samples and validating SAL-produced LSD spikes.
- Provide a back-up functionality in case of a planned or emergency shut-down at SAL. This implies logistical support, such as sample receiving, tracking, splitting, storage and distribution.
- For the case of planned or unplanned shut-down of SAL, back-up capacity for the analysis of 800~1000 nuclear samples per year needs to be allocated.

## 3 Problems of an Ageing Laboratory

### 3.1. Conclusions of the SAL Workshop[1]

The International Atomic Energy Agency convened a meeting of technical experts in Seibersdorf, Austria, in November 2006 to help prepare SAL to meet the future needs of the Department of Safeguards. Critical components of SAL's nuclear laboratory are nearing the end of their design lifetime and the IAEA must respond soon with an appropriate plan for facility infrastructure improvement, either by renovating the existing structure or by building a new laboratory. The decision to renovate the nuclear laboratory or build a new structure is not straightforward. Each option involves a significant financial investment. Further, safeguards regimes will likely change and therefore may require new types of analytical services and facilities. Therefore, the capabilities required of SAL must be evaluated from the perspective of the projected needs of the Department of Safeguards 10 to 20 years in the future.

The workshop participants addressed SAL's present and future role in meeting the needs of the Department of Safeguards. First, the group agreed that SAL fulfills the present day needs of the Department of Safeguards. The group then considered three alternatives:

1. Consolidating all analytical services at SAL
2. Eliminating SAL and distributing work among a more developed laboratory network
3. Optimizing the distribution of work between SAL and NWAL

The first option was rejected because it was deemed vulnerable to a single point of failure. The group rejected the second option due to skepticism over the feasibility of establishing a distributed laboratory network and because the analytical capabilities might not fall under sufficient control of the IAEA. The third option was deemed the most reasonable. This option was chosen because it would enable a core capability to be maintained under full IAEA control while providing for back-up and surplus capacity. The participants agreed that SAL should be the central laboratory (strongly supported by the NWAL to the extent feasible) in providing analytical support services to the Department of Safeguards. Although support from NWAL for environmental sample analysis was found to be adequate, nuclear materials analysis support was judged to be underutilized. The participants agreed that there was the potential for more support within Member States and that an expanded NWAL should be further investigated.
The workshop participants therefore recommended that the IAEA determine the realistic capabilities and capacity of the NWAL for nuclear materials analysis. The IAEA should then conduct a cost benefit analysis, taking into account quality and timeliness of results, and cost. One of the main conclusions was that taking no action is not considered a viable option.

The following recommendations have been formulated by the workshop participants:

1. The IAEA should develop a comprehensive project plan that addresses: engineering studies, optimization of NWAL, legal issues, staffing, continuity of expertise, and optimization of space in the existing SAL.
2. The NWAL should be strengthened by qualifying additional laboratories for the NWAL. The IAEA should invite Member States to express their capabilities and commitment to provide analytical services to the IAEA as part of the NWAL. The agency should consider placing a fixed deadline on the communication of Member States responses.
3. The IAEA should perform a feasibility study and cost analysis for the options of renovating, rebuilding, or relocating the nuclear material laboratory. The study should include an analysis of the optimum workload distribution between SAL and NWAL.
4. The timeliness requirements for performing sensitive particle analysis should be reviewed and revised as appropriate. The Department of Safeguards should assess the impact of a one month reporting requirement and then determine the best approach for meeting the timeliness criteria.

## 3.2. Current Problem Areas

The TUeV-*Report on the Sustainability of Operations at the Safeguards Analytical Laboratory* (included in the Workshop report) states unambiguously that problems are to be expected in the near future with the ventilation system to occur more frequently. The probability of prolonged outages will thus increase gradually over the years to come. Though no immediate catastrophic failure is foreseen, urgent action is required in addressing the concerns to ensure sustainable operations in the near future.

*From a technical point of view, only one of the three exhaust air ventilators in the central glovebox exhaust air system is in a fully acceptable condition. Corrective action is urgently needed. ……*
*In the years to come, we judge that the number of irreparable faults in the exhaust air systems and filter housings will increase. The number and duration of interruptions in laboratory operation will likewise increase. Since the individual exhaust air systems are connected with all the laboratory rooms of their zone via collecting ducts, the failure, for example, of a complete central exhaust air system, e.g. a radionuclide fume hood exhaust air system, will affect all the laboratory rooms in that zone. Obtaining spare parts for the ventilation systems is already proving problematic and the problems will get worse in the future. Unplanned replacement of old components as they fail (e.g. old HEPA filter housings) with new ones involves a lot of work and takes up a lot of time, because ventilation ducts need to be modified and account needs to be taken of delivery times. Interruptions in operation lasting several weeks are to be expected. Therefore, corrective action is urgently needed.*[2]

This evaluation is substantiated by findings of the immediate past: whereas previously routine maintenance was sufficient to ensure lab operations, in the past two years major components had to

be replaced (steam injectors, control valves, main exhaust ventilator) – with associated major problems resulting from the precarious spare part situation.

The entirety of SAL suffers from severe space limitations. Owing to this severe lack of space and to the necessity to perform multiple operations in restricted laboratory space (see figure 3), there are increased risks to the safe handling of nuclear and radioactive materials. In addition, though much effort has been invested to upgrade safety and security at the nuclear laboratory, the facility located in rented space is not fully compliant with the relevant international safety and security standards and it is unlikely that it can be made compliant without significant expenditure. And finally, due to multiple extensions and additional rented space at the campus, the patchwork of rooms and laboratories does not provide the logistical background for efficient sample analysis.

Figure 3: Heavy space constraints result in multiple activities taking place in one room, causing higher-than-necessary exposures to staff

Furthermore, security at SAL deserves critical attention. United Nations Security Standards have been drastically upgraded and SAL residing in rented space is not in a position to fully meet these standards. Similarly, a facility handling substantial amounts of fissile material needs to be in compliance with the Convention on Physical Protection of Nuclear Material (CPPNM). However, the ARC campus has relaxed its security efforts considerably after decommissioning of the old research reactor and the elimination of their needs of compliance with CPPNM.

And lastly, equipment issues. SAL needs to replace obsolete equipment on a regular basis, or at least before equipment falls in disrepair because of spare part unavailability. This effort alone amounts to approximately 1 million Euro per year – money which has not been invested over the previous years – and thus has led to a precarious state of service availability.

It has been argued that older laboratories of similar profile exist and do not need expensive total renovation. This is a very serious argument worth addressing.

- SAL is primarily a service provider operating at full capacity. Timeliness is of essence.
- Some laboratories are considerably larger, allowing an incremental approach to renovation.
- SAL does not have separate ventilation circuits, allowing partial shut-down.
- Any kind of major renovation work at SAL will invalidate the “grandfather principle”.

### 3.3. Options to Address the Infrastructure Problem

In line with workshop recommendations, a cost-benefit analysis has been initiated considering all options for upgrading SAL to address the problems of the technical infrastructure as well as space and security concerns. Though some options have already been identified in the workshop, attention has to be given to many details such as legal background, changes in applicable rules and regulations, etc. Therefore, only preliminary findings may be provided, possibly interesting for further discussion.

In view of the current status of SAL, taking no action to upgrade its infrastructure is not considered to be a viable option. However, properly addressing infrastructure concerns will require significant capital investment. Certain issues are specifically cost sensitive and need careful consideration:

- The current contract for lease of the facility does not foresee decommissioning at the expense of the Agency when the premises are vacated. However, any renovation of the facility will incur substantial decommissioning costs;
- In case of renovation, the Austrian authority for licensing has confirmed to require compliance with current building codes (e.g. fire protection) and radiation safety standards;
- Since there is insufficient capacity in the NWAL to analyze all nuclear material samples during SAL's shut-down for renovation, consideration would need to be given to the possibility of commercial analysis of such samples at potentially high cost (this may create implications regarding confidentiality that would need to be resolved); and
- Security upgrades at the present location to ensure compliance with current standards may be possible but would certainly entail significant, additional costs.

Three main options regarding upgrading the infrastructure have been considered: renovation of the existing facility, construction of a new laboratory and a hybrid solution containing elements of both. Main elements in the consideration of these options are discussed below.

- Renovation of the laboratory would involve its complete renewal at its current location after decommissioning. The overall process might take up to two years and would not adequately address the severe space limitations and security concerns. This option would not satisfy current or future analytical requirements.
- Rebuilding the laboratory within the future secure complex of the Agency's Laboratories at Seibersdorf would provide for the most comprehensive solution by addressing all safety and security issues as well as space requirements. There would be no decommissioning costs for the IAEA and SAL would be able to continue to operate at nominal capacity.
- A hybrid solution would consist of a renovated SAL serving as nuclear laboratory in its entirety and an additional office and (non-nuclear) laboratory facility elsewhere that would meet the space requirements. However, this option would share many of the drawbacks of the renovation option.

A preliminary comparative assessment cannot help to take note of the unexpected cost factors involved in renovation as well as a number of non-dollar (or non-Euro) items such as compliance with security standards. Though this may be considered a negotiable item, the liabilities incurred need to be carefully evaluated.

### 3.3. Timelines

The SAL Workshop had recommended a feasibility analysis, taking into account all options and a cost-benefit-analysis. This analysis is under way, but has not been completed yet. Beyond undeniable infrastructure facts policy decisions are involved. How much independent analysis capability does the Agency need to credibly validate Safeguards conclusions? What is the best load sharing between SAL and NWAL? Currently, these issues are being seriously addressed to comply with the recommendations made at the Workshop. As of now, it would be premature to foresee the end of these deliberations, possibly before the end of this year.

Though no urgent and immediate action is required, a decision on a Plan of Action has to be established soon. The implementation of any kind of decision to address the infrastructure problems is likely to take considerable time, while the risk of failure of the ageing components is increasing. It would be wise to plan for a completion of the chosen option by the end of 2012 (after approximately five years).

## 4 Conclusions

The Agency has embarked on a process to develop a Plan of Action to address the infrastructure problems at SAL. No results are available to date. Based on workshop recommendations, it is likely that the Agency will endorse the recommendation to have a strong in-house capability for nuclear and environmental sample analysis, strongly supported by a network of analytical laboratories. IAEA analytical capability is perceived necessary to independently validate any kind of sample analysis conducted in the network, whereas reasonable IAEA analysis capacity needs to be foreseen to allow SAL and NWAL meet timeliness expectations. Whereas environmental sample analysis already demonstrates good network utilisation, nuclear material analysis is focused at SAL currently. A stronger participation of qualified nuclear laboratories in NWAL seems desirable. Since overall nuclear sample analysis demand is perceived as basically stable for the foreseeable future, the Agency's Safeguards Analytical Laboratory is likely to be upgraded to ensure sustainability of operations in the mid- to long-term future.

Analytical techniques for nuclear material analysis have not been the object of this discussion. Fulfilling services to the Department of Safeguards primarily requires techniques compliant with requested analyses. Though an evolution of employed techniques can be observed, basic requirements are foreseen similarly stable as predicted sample numbers. Additional requirements for destructive analysis (such as impurity analysis) are discussed elsewhere during this meeting, concerning SAL infrastructure issues it has been observed that additional techniques do not generally make traditional methods obsolete – thus resulting in additional needs on floorspace. The current feasibility study seems well advised to allow for a reasonable growth potential if the implementation should last for another 30 years.

SAL has served the Agency well over the previous 30 years, adapting to numerous challenges. Though results of recommended analyses on feasibility and cost-benefit of various options are not available yet, it seems probable that SAL – possibly upgraded – will continue to have a significant part in the future of nuclear material analysis of the IAEA.

[1] The Safeguards Analytical Laboratory: 2010 and Beyond (Report of the SAL Workshop), Feb 19th, 2007, available from the International Atomic Energy Agency

[2] . TUeV-*Report on the Sustainability of Operations at the Safeguards Analytical Laboratory* (included in Workshop report[1]), section 5.3 [Containment of radioactive material and ventilation facilities]

# Development of uranium isotopic reference materials at IRMM

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## *Abstract*

Over the last 5 years, IRMM has been developing new sets of isotope reference materials based on proven methods of purifying and mixing highly enriched oxides. In the first stage, oxides of $^{233}U$, $^{235}U$ and $^{238}U$ were purified and from these a set of 10 mixtures were made in which the $^{235}U$:$^{238}U$ ratios was 1:1 and the $^{233}U$ abundance varied from 0.3 to $10^{-6}$ (IRMM-074). From the solutions of $^{235}U$ and $^{238}U$ a number of mixes (20%, 35%, 75%, 90% $^{235}U$ enrichment) were also made to act as primary isotopic mixes for later calibration and certification of uranium isotopic standards jointly with DOE/NBL.
Two further uranium materials were purified: a natural uranium and an enriched $^{236}U$ oxide. From these materials a set of isotopic standards of $^{236}U$ in natural uranium, with $^{236}U$ abundances from $10^{-5}$ to $10^{-9}$ were made and certified. These were then verified against the new series IRMM -074.
From the $^{233}U$ and $^{236}U$ solutions a 1:1 spike has been made and certified. The advantage of this material was shown in the certification of the mixtures of $^{235}U$ and $^{238}U$ to measure the mass-fractionation on each filament measured by TIMS.

The purification, mixing and certification of the new materials will be discussed and the application of these new reference materials in measurements of uranium by mass-spectrometry will be illustrated.

**Keywords:**
Isotopic reference materials; uranium, mass-spectrometry

## *Introduction*

The Nuclear Materials Reference Centre of the European Commission at IRMM, Geel has been involved in the development of isotopic reference materials suitable for nuclear materials accounting and safeguards since its inception over 40 years ago. Several series of isotopic reference materials (IRMs) for uranium were produced and certified, either in the gas phase or as solutions in nitric acid [1]. There is a continual need for new isotopic reference materials, however, with the highest specifications to match the needs of modern measurement techniques.
As the stocks of the well known series IRMM-072 became low it was decided to restart the isotope mixing program applying modern techniques such as clean-room facilities to control cross-contamination effects. The basic principles used in making previous isotopic reference materials were adhered to as these had proven to be very successful. It was also seen as an opportunity to apply modern metrological principles to the certification, especially with regard to the specification of the uncertainties of each mixture.

## *Needs for uranium isotopic reference materials*

Isotopic mixtures, in which isotopic ratios are certified, are applied extensively in the measurement of uranium materials throughout the nuclear fuel cycle as well as in measurements of environmental samples. The needs in the latter case can be quite different from those in conventional material accountancy.
The reference materials are applied to calibrate the instruments and also to ensure that the isotopic ratio measurements are being carried out in a controlled fashion: that is in a quality assurance role.
Reference materials in which the isotopic content is certified are applied differently in practice. These are the basis of measurements of amounts of nuclear material, usually applying isotope dilution mass-spectrometry.
In nuclear material accountancy, certified isotopic reference materials of $^{235}U$ and $^{238}U$ ranging from depleted through to enriched in $^{235}U$ are a corner-stone of modern mass-spectrometry measurements for this element. Except for depleted to low enriched uranium, there is a lack of IRMs with a sufficiently low uncertainty.
Triple isotopic mixtures, as for instance typified by the IRMM-072 series [1] are applied for specific purposes: in this case to demonstrate the linearity of the instrument in measuring isotopic ratios over 6 orders of magnitude.
Recently, a need has occurred for mixtures containing $^{236}U$, which is an isotope that because of the very large range in abundances (from natural uranium with an abundance in the order of $10^{-11}$ or lower through to uranium from reprocessing nuclear fuel, in which the abundance can be a $10^{-2}$ or higher) places an extreme challenge to the measurement laboratory.
Other special needs include spikes with $^{233}U$ together with $^{236}U$ and a 1:1:1:1 isotopic mixture with $^{233}U$, $^{235}U$, $^{236}U$ and $^{238}U$.
All these needs were kept in mind in the design and organization of the uranium isotope mixing program carried out over the last few years.

## *Principles of mixing uranium oxides*

The primary material in the whole of the analysis system is uranium metal: pure metals are available for purchase for natural and uranium enriched in $^{235}U$, but not for other isotopes, e.g. $^{233}U$, or $^{236}U$. However it was realized in the development of isotopic mixtures carried out earlier at IRMM that careful control of the conditions could permit pure, enriched isotopes to be mixed together in the form of oxides. This was the principle employed for the $^{235}U/^{238}U$ mixtures certified at IRMM previously.
The method employed is as follows:
Highly enriched isotopic uranium is purchased. Each material is chemically purified using the same method and chemicals and finally prepared in the form of $U_3O_8$ by sintering at between 900 and 950 C under controlled humidity (< 50 ppm $H_2O$). The oxides are then carefully weighed and dissolved, either together or separately.
This method allows isotopic mixtures to be made with ratios having very low combined uncertainties. Larger uncertainties may be expected for the individual isotopic contents of course.

## *First sets of new standards*

Initial internal tests were carried out at IRMM to determine if an improvement in the chemical purification methods was feasible. The availability of ICP mass-

spectrometers to readily measure the impurity levels in solutions after each step was a big advantage. The method finally employed was in fact a classical one: purification in nitric acid medium on an anion exchange column, then on a cation column and finally by precipitation as peroxide and heat treatment to the final oxide stage. This method brought the final level of total impurities to below 50 ppm (a value that is input as a conservative 100 ppm in the calculation of total uncertainty). Further purification steps did not reduce the level of impurities.
A new glove-box was constructed to allow the final, pure oxides to be sintered in an oven at between 900 and 950 C in a controlled, low-humidity atmosphere. The water content in the glove box was held below 50 ppm to ensure the stability of the oxides after sintering. An ancillary glove box was used for the mass-metrology and to store the oxides: this was also kept under the same low-humidity atmosphere as the box with the oven.

## IRMM-074

The test of the method, brought up to date but based on the original methods of 30 years ago, was the production of isotope mixtures of $^{233}U$, $^{235}U$ and $^{238}U$ in a replacement set, IRMM-074, for IRMM-072 [2]. The three oxides were sintered in the same oven and $^{235}U$ and $^{238}U$ oxides were dissolved together in nitric acid to make a solution with an isotopic ratio 1:1( IRMM-3050); the $^{233}U$ oxide was dissolved in nitric acid at the same time (IRMM-3630). The IRMM-3630 solution was then diluted to provide 3 further solutions that were then mixed by weight with IRMM-3050 to make the series IRMM-074. This has 10 components, all having the same ratio $n(^{235}U)/n(^{238}U)$ that originates from IRMM-3050 but with a concentration of $^{233}U$ that varies stepwise from a ratio 1:1 $n(^{233}U)/n(^{238}U)$ down to $10^{-6}$:1.
The $n(^{235}U)/n(^{238}U)$ ratio in IRMM-074 was verified by careful measurements applying TIMS to this material and two others - IRMM-199 and IRMM-072. The measured value, using the other two materials to calibrate the method, gave an excellent agreement with the certified value of the ratio based on the masses of the oxides and solutions.
The $^{233}U$ contents of the first 7 IRMM-074 mixtures were also verified by TIMS using Faraday collectors [3]. The mass-bias was controlled by measuring the $n(^{235}U)/n(^{238}U)$ ratio in each sample. The results are shown in Figure 1. The excellent agreement verifies not only the certification $^{235}U/^{238}U$ mixture but also the concentration of the solution containing the $^{233}U$: IRMM-3630.

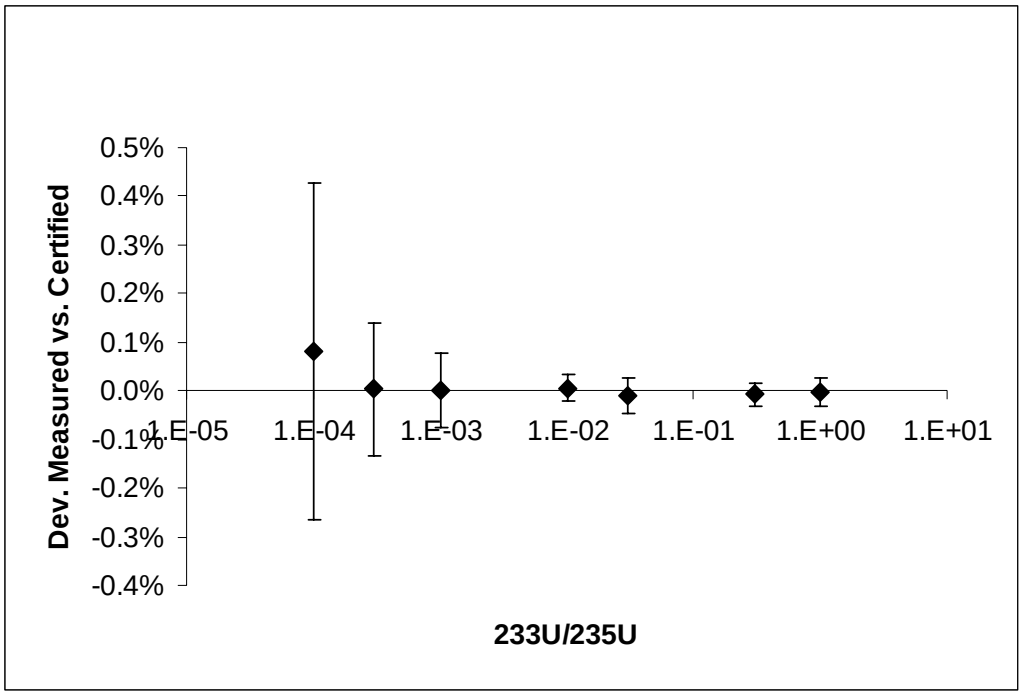


**Figure 1: Verification by TIMS of $^{233}U/^{235}U$ ratio in IRMM-074**

## IRMM-075

The measurement of $^{236}U$ has demonstrated some weaknesses in the capability of the international community as demonstrated in recent inter-laboratory comparison exercises. Measurement of this isotope is difficult because it has a very wide dynamic range, sits on the tail of the much larger intensity $^{238}U$ peak and because there have not been isotopic standards available for this isotope to allow laboratories to calibrate and check their measurements.

Following the proof of the method in IRMM-074, a new series with two components, natural uranium and $^{236}U$ was started based on the same principles. A natural uranium with no detectable $^{236}U$ content by TIMS was chosen. The $^{236}U$ abundance in the natural material was subsequently measured by accelerator mass-spectrometry (AMS) and shown to be ca. $5 \cdot 10^{-11}$. A sample of enriched $^{236}U$ was provided by the IAEA, Vienna. Both materials were then chemically purified in separate, new glove-boxes, applying the methods used for IRMM-074. The two oxides were then sintered in parallel under the same conditions and dissolved to make 2 solutions in nitric acid. The solution of $^{236}U$, (IRMM 3660), was diluted to make 5 further solutions with lower concentrations of uranium. All solutions were kept as standard in cleaned quartzware.

From the solution of natural uranium (IRMM-3001) and the solutions of $^{236}U$ a series of mixtures were made with abundances of $^{236}U$ down to $10^{-9}$. These form a series: IRMM-075 [4].

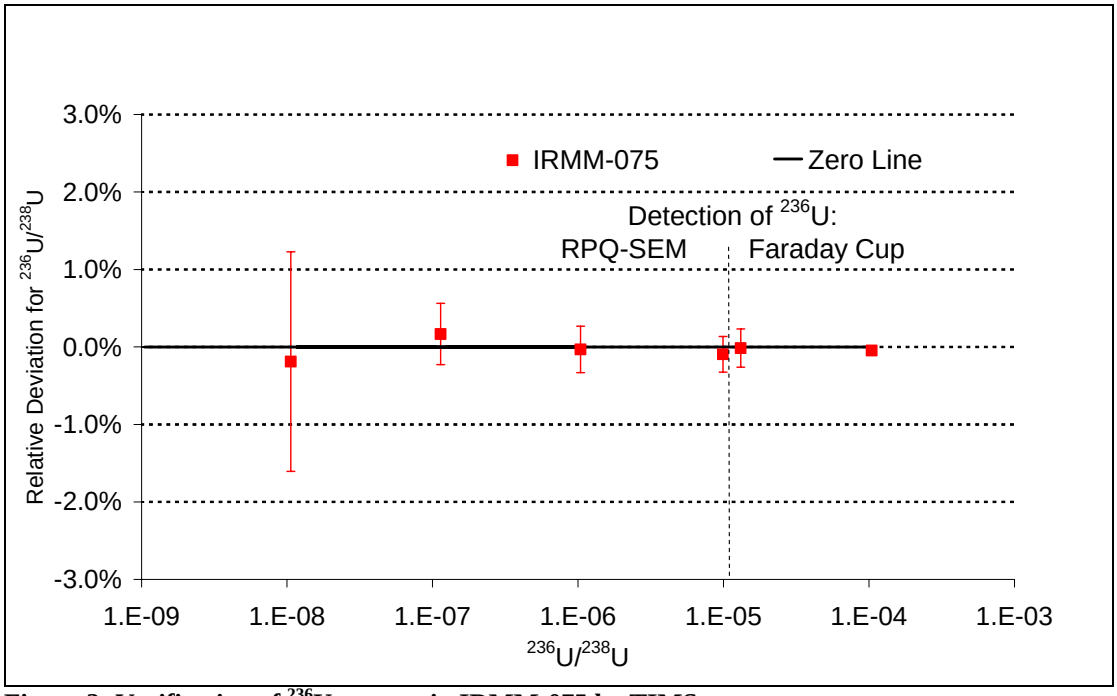


**Figure 2: Verification of $^{236}U$ content in IRMM-075 by TIMS**

## $^{235}U/^{238}U$ isotope mixtures

Part of the uranium isotope mixing program has been to also produce primary isotopic mixtures of $^{235}U$ and $^{238}U$ across the complete range to be applied for the eventual certification of an international series of uranium isotopic reference materials together with the New Brunswick Laboratory, USA. The original $^{235}U$ and $^{238}U$ oxide materials were dissolved individually into separate solutions in nitric acid, which allowed them to be mixed to form isotopic mixtures with 20%, 35%, 75% and 90% abundances of $^{235}U$. Together with the 50% mixture (IRMM-3050) applied for the IRMM-074 mixtures they will form a basic set that can be used to calibrate measurements and therefore certify other isotope mixtures.

## Isotope mixtures with $^{236}U$

The basic solution of enriched $^{236}U$ will be used for three goals: as a $^{236}U$ spike, as a 1:1 spike mixture with $^{233}U$ and in an isotopic mixture with $^{233}U$, $^{235}U$ and $^{238}U$ to yield a solution with ratios of 1:1:1:1 of these isotopes. Both the $^{233}U/^{236}U$ mixture (IRMM-3636) and the mixture with 4 isotopes (IRMM-3100) have the potential to be used as spikes as well as isotopic reference materials.

## *Present situation*

The mixing program for the uranium isotopes has proven to be complicated and has demanded a lot of time. No mixture is assumed to be known completely, and cannot be certified, until verification measurements are carried out. Modern thermal ionization mass-spectrometry is capable of measuring isotope ratios with a reproducibility of 0.05% or lower and the original $^{235}U/^{238}U$ isotope mixture, (IRMM-3050, the base for the IRMM-074 series) is a strong foundation as calibrant for comparisons, as this mixture has been verified to a high degree against other existing high quality isotopic reference materials as described above.

The series IRMM-074 and IRMM-075 are completed and certified. The other mixtures are not yet at this stage. Many comparison measurements are presently being carried out and these will not be certified until later in 2007 at the earliest.
The prospect of having new, independently produced and certified isotope mixtures, with, for the first time, certificates based fully on the GUM will help raise the level of this class of reference materials and make them available worldwide.
It will be particularly interesting to see how well the isotope content as well as the isotope ratios can be certified for the $^{233}U/^{236}U$ mixture and also the 1:1:1:1 isotope mixture. It is conceivable that if the certified uncertainties of the isotope content are very low, new applications will be found for these materials leading to extremely low uncertainties in the measurement of the isotopic content of uranium in real samples.
A certified ratio as well as isotope content of a $^{233}U/\ ^{236}U$ mixture will allow the mass-fraction during a measurement to be calibrated and am isotope dilution measurement to be made at the same time. This will, in theory, permit highly accurate IDMS measurements to be carried out. Such a double spike may find application in the measurement of uranium in environmental samples where the sample size is limited.

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# A New Set of Uranium Isotope Reference Materials for the Improvement of Nuclear Safeguards in South America

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***ABSTRACT***

*Uranium isotope reference materials are essential tools to provide accurate and therefore reliable measurement results of nuclear materials.*

*In order to improve the trust in the nuclear materials safeguards system in South America, ten materials in the form of uranium hexafluoride with isotopes amount ratios ranging from 0.5 to 20.0 % of $^{235}U$ in mass were recently prepared. They were enriched, purified and characterized in the Brazilian laboratories but the isotopic measurements were performed in the Institute of Reference Materials and Measurements. Modern concepts of metrology in chemical measurement were employed throughout to provide reference materials with certified values traceable to the SI and having the lowest achievable uncertainties.*

***Keywords**:* mass spectrometry, metrology in chemical measurement, nuclear safeguards, uranium isotope reference materials

# 1. INTRODUCTION

Modern mass spectrometry is continually enhancing its capability to provide isotope amount ratio measurement results with higher repeatability, often relying on smaller sample amounts.

This is not enough, however, to guarantee the accuracy of measurement results because there is always some kind of bias affecting the measurement process. The mass discrimination is regarded as the most important factor responsible for this bias.

The experimental values obtained are therefore corrected for mass bias by measuring certified isotope reference materials under the same instrumental conditions that were used to measure the samples.

A certified reference material is defined as a "reference material, accompanied by a certificate, one or more of whose property values are certified by a procedure which establishes traceability to an accurate realization of the unit in which the property values are expressed, and for which each certificate value is accompanied by an uncertainty at a stated level of confidence [1].

Uranium isotope reference materials have being made available to the nuclear analytical community mainly from two very well known producers, the New Brunswick Laboratory (NBL) (Chicago, USA) [2] and the Institute for Reference Materials and Measurements (IRMM) (Geel, Belgium) [3]. These two laboratories have not just the expertise but also the most advanced facilities and analytical instrumentation to produce these materials.

In spite of this, field laboratories in South America face some other hardships to accomplish their mission to provide accurate isotope ratio measurements results.

First, commercial isotope reference materials in the form of $UF_6$ can only be obtained in the western world from IRMM and with enrichment levels limited to 4.5 % in $^{235}U$ in mass.

Second, the increasing barriers presently imposed on the transportation of radioactive materials over international borders complicate the acquisition process of these essential materials.

These arguments led to the establishment of a scientific programme focused on the preparation, characterization and certification of isotope reference materials [4] under the modern concepts and practices of metrology in chemical measurement [5, 6, 7 8].

The purposes of this programme are twofold: to identify the most critical factors in the preparation and certification of these materials and to provide the Brazilian laboratories with a set of reference materials in the range of 0.5 to 20.0 % $^{235}U$ in mass needed to perform accurate uranium isotope ratio measurements.

# 2. Metrological concepts

Programmes aimed to the production of certified reference materials must necessarily be founded on sound metrological concepts because they bring consistency, quality and transparency to the certified values carried by these materials.

Metrology is defined as the "science of measurement" including all aspects both theoretical and practical with reference to measurements, wherever their uncertainty, and in whatever fields of science or technology they occur [6].

Some of the metrological concepts applied in this programme are described below.

### 2.1 Mesurand

The mesurand is defined as "particular quantity subjected to measurement" [6]. In this programme the isotope amount ratios, namely $n(^{234}U)/n(^{238}U)$, $n(^{235}U)/n(^{238}U)$ and $n(^{236}U)/n(^{238}U)$, are considered the mesurands.

### 2.2 Traceability

Traceability is defined as the "property of the result of a measurement or the value of a standard whereby it can be related to stated references, usually national or internationally standards, through an unbroken chain of comparisons all having declared uncertainties" [6].

The traceability of the samples to the International Systems of Units (SI) was established selecting reference materials that have the closest link to primary reference materials standing at the top of the metrological traceability chain. These materials were produced by the mixture of highly enriched uranium oxides [9, 10].

The realization of the traceability was accomplished in practice by the comparison of the isotope ratio values provided by reference materials and samples, as required by the analytical procedure.

The isotope reference materials used for this purpose are described in the tables below. The expanded uncertainty values are presented in parenthesis ($k$=2) after the values of the ratios.

| Certified reference materials | $n(^{235}U)/n(^{238}U)$ |
|---|---|
| IRMM 031 | 0.0032157 (16) |
| IRMM 071 | 0.0072623 (22) |
| IRMM 194 | 0.0200552 (60) |
| IRMM 295 | 0.0307711 (92) |
| IRMM 446 | 0.0473245 (14) |

**Table 1** Isotope amount ratio of IRMM certified reference materials

| Certified reference materials | $n(^{235}U)/n(^{238}U)$ |
|---|---|
| NBL U 100 | 0.11360 (11) |
| NBL U 150 | 0.18109 (18) |
| NBL U 200 | 0.25126 (26) |

**Table 2** Isotope amount ratio of NBL certified reference materials

### 2.3. Uncertainty estimation

The uncertainty values associated with the measured isotope amount ratios were obtained using the ISO-GUM guide [7], a comprehensive, standardized and transparent procedure for uncertainty estimation. Moreover, for practical reasons, the software GUM Workbench [11] was used to speed-up the calculations.

## 3. Experimental

### 3.1 Material preparation

Uranium hexafluoride ($UF_6$) was enriched to produce ten base materials with isotope ratios ranging from 0.5 to 20.0 % $^{235}U$ in mass**.**

Storage ampoules type 1S and 3S manufactured in Monel and in stainless steel respectively were cleaned, helium leak tested and conditioned with fluorine gas ($F_2$) prior to receiving the $UF_6$ samples.

Approximately 300 g of each of these base materials were distilled into ampoules 1S. They were kept at a temperature of - 80 $^0$C in a bath of acetone and carbon dioxide to allow the light gases such as $H_2$, $N_2$, $O_2$ and HF to be pumped off.

The ampoules were then heated for 1 h at 150 $^0$C, allowing the $UF_6$ to liquefy. They were also submitted to a vigorous shaking to provide a thorough chemical and isotopic homogenisation.

A first sub-sample with 3.0 g of $UF_6$ was cryogenically transferred from each ampoule 1S to an ampoule 3S to allow the isotope amount ratio measurement by gas source mass spectrometry (GSMS).

A second sub sample with 0.5 g of $UF_6$ was transferred cryogenically to a glass ampoule installed in a sampling system and kept a temperature of -196 $^0$C by the immersion in liquid nitrogen.

Then, 30 mL of deionised water (18 MΩ·cm) was introduced in the glass ampoule containing $UF_6$ to generate uranyl fluoride according to the reaction below:

$$UF_6 + 2\ H_2O = UO_2F_2 + 4\ HF \qquad (1)$$

The uranyl fluoride solution was dried in a hot plate kept for 12 h at 60 $^o$C to remove the existing HF. Nitric acid Suprapur 65 % 8M, manufactured by Merck (Darmstadt, Germany), was added to allow the formation of uranyl nitrate according to the following reaction:

$$UO_2F_2 + 2\ HNO_3 = UO_2\ (NO_3)_2 \cdot H_2O + 2\ HF \qquad (2)$$

The uranyl nitrate solution was then kept in a muffle for 1 h at 900 $^o$C, enabling the formation of octauranium trioxide, as indicated below:

$$UO_2\ (NO_3)_2 \cdot H_2O = U_3O_8 \qquad (3)$$

The $U_3O_8$ produced was weighed, dissolved by the addition of nitric acid Suprapur 65% 8M and purified in an ion extraction column Dowex 1 x 4, 100-200 mesh manufactured by Dow Chemical (Midland, MI, USA). The eluted uranium solution was carefully dried in a hot plate.

The resulting oxide was redissolved with nitric acid generating $UO_2\ (NO_3)_2$. The solution concentration was finally adjusted to 5.0 mgU/mL, value required to run the isotope ratio measurements by thermal ionisation mass spectrometry (TIMS).

## 3.2 Chemical characterization

The chemical characterization of the samples consisted in the measurement of the volatile and non-volatile compounds that may be considered as impurities in $UF_6$ [12]. This step is important because the materials produced can be used as a reference for both chemical purity and isotopic measurements [13].

The most typical volatile impurities (HF, $CoF_2$, $BF_3$, $CF_4$, $SiF_4$, $PF_5$, $SF_6$ and $WF_6$) were measured by the Fourier Transformed Infrared Spectrometry (FITR) technique that allows the detection of such impurities at mg/gU levels [13].

To perform this measurement, a FTIR spectrometer, model 1750, manufactured by Perkin Elmer (Norwalk, CT, USA), was coupled to a leak tight stainless steel cell assembled with AgCl windows where the $UF_6$ samples to be measured were trapped.

The non-volatile compounds (twenty-four elements) were measured by inductively coupled plasma mass spectrometry (ICPMS), capable of measuring elemental concentrations at µg/gU levels[14].

An ICPMS mass spectrometer, model PQII, manufactured by Fisions (Winsford, Cheshire, England) associated to the matrix matching method was employed to perform this measurement [15].

### 3.3 Isotopic characterization

The isotope characterization was the most relevant task in this programme because the quantities to be certified were the isotope amount ratios $n(^{235}U)/n(^{238}U)$, $n(^{234}U)/n(^{238}U)$ and $n(^{236}U)/n(^{238}U)$. These data eventually allowed the calculation of the molar and the mass fraction of each sample.

As one of the main goals was the measurement of isotope ratios with the lowest achievable uncertainties, the instrumental parameters that provided the highest signal intensities in the spectrometer's ion detectors were always selected. Thus, whenever possible, Faraday detectors were chosen because they allow ion current measurements with better accuracy than the secondary electron multiplier (SEM) device.

The isotope amount ratio $n(^{235}U)/n(^{238}U)$ was measured using a MAT 511, an electron impact mass spectrometer manufactured by Varian MAT (Bremen, Germany). It is equipped with a $90^{o}$ magnetic sector analyser and two fixed Faraday collectors to measure the ratio of the two major isotopes in $UF_6$ samples.

The measurements were carried out using the double standard method, which relies in the bracketing of the sample by two isotope reference materials (IRM). The first IRM had an isotopic ratio slightly higher and the second one an isotopic ratio slightly lower than the sample[16].

The isotope amount ratios $n(^{234}U)/n(^{238}U)$ and $n(^{236}U)/n(^{238}U)$ were measured using the Triton, a thermal ionisation mass spectrometer manufactured by Thermo Electron (Bremen, Germany). It is equipped with a sample magazine for twenty-one filaments, $90^{o}$ magnetic sector analyser, dynamic zoom optics and nine Faraday collectors, each one associated with its own signal amplifier.

For small signals the Triton also has a SEM device in combination with a retarded potential quadrupole (RPQ). This serves as an energy filter and reduces the contribution from the large $^{238}U$ ion beam to the much smaller $^{236}U$ ion beam.

The measurements were carried out mostly at the intensity of 10 V, using the modified total evaporation method in the static mode [17]. Both sample and reference materials were processed using the same operational parameters.

The measurement of the minor isotope ratios for some samples of the set employed a different method [18], where the $^{238}U$ ion beam was kept at the highest possible intensity, typically 30V.

In the samples where the isotope ratio $n(^{236}U)/n(^{238}U)$ was smaller than $10^{-5}$, the SEM device was employed. The $^{234}U$ ion beam was then used to run the inter-calibration routine between the SEM and the Faraday multi-collector.

The GSMS and TIMS techniques described above were selected to be employed in this programme because comparative studies with other techniques demonstrated they were capable to provide the smallest achievable uncertainties for uranium [19].

## 4. Results and discussion

### 4.1 Chemical characterization results

The measured concentrations of volatile impurities are presented in table 3 for a group of three samples, representative of the categories of natural, enriched and depleted materials in $^{235}U$.

| Volatile impurities | MRI 0.5 | MRI 0.7 | MRI 2.5 |
|---|---|---|---|
| | mg/100g | mg/100g | mg/100g |
| **HF** | 4.7 ± 0.5 | 4.5 ± 0.5 | 4.6 ± 0.5 |
| **$CF_4$** | 0.004 ± 0.002 | 0.003 ± 0.002 | 0.006 ± 0.002 |
| **$SiF_4$** | 0.003 ± 0.002 | 0.004 ± 0.002 | 0.005 ± 0.002 |

**Table 3** Concentrations of volatile impurities in $UF_6$ samples

The volatile impurity detected in the highest concentration was the HF, formed by the decomposition of $UF_6$ in the presence of air humidity. This gas can be easily removed by high vacuum pumping if the ampoules are kept at temperatures below - 80 $^0$C.

The infrared spectrum of sample MRI 2.5 is presented in figure 1. The wavelength ($cm^{-1}$) is presented in the abscissa, while the transmittance (T%) is in the ordinate. The peaks featured at the right of the spectrum are those associated with the $UF_6$ (677 $cm^{-1}$) while the peaks at the left are associated with the HF (4 039 $cm^{-1}$). No other impurities were identified in this spectrum.

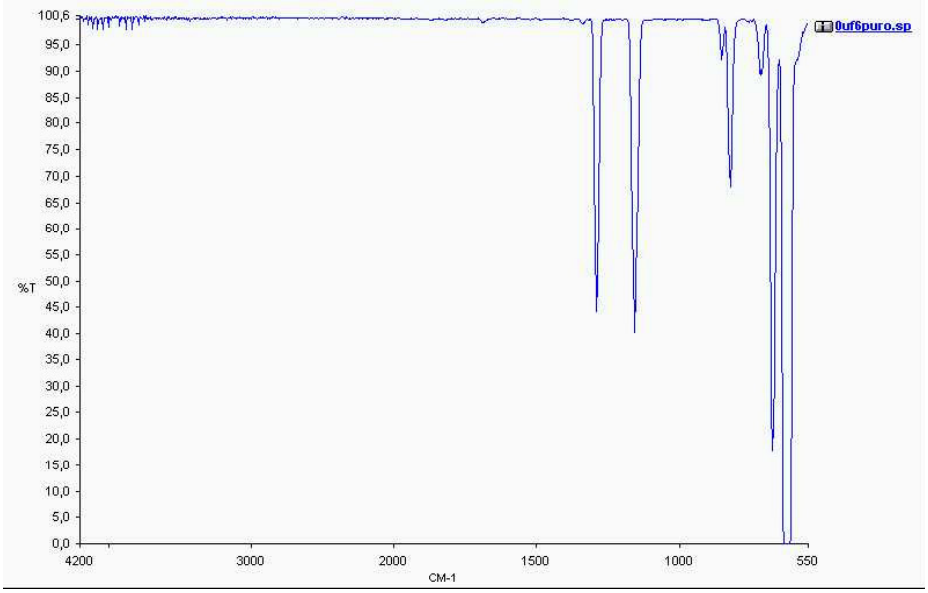



**Figure 1** Spectrum of sample MRI 2.5 measured by Fourier Transformed Infrared Spectrometry (FTIR)

The sum of the measured concentration of twenty-four elements (Be, B, Na, Mg, Al, Si, Ca, Ti, V, Cr, Mn, Fe, Co, Ni, Cu, Zn, Zr, Mo, Ag, Cd, Sn, W, Pb, Bi) plus eighteen rare-earth elements (Sc, Y, La, Ce, Pr, Nd, Sm, Eu, Gd, Tb, Dy, Ho, Er, Tm, Yb, Lu, Hf and Th) is presented in table 4.

| Non-volatile impurities | MRI 0.5 | MRI 0.7 | MRI 2.5 |
|---|---|---|---|
| | µg/g | µg/g | µg/g |
| Total | 152 ± 24 | 158 ± 24 | 185 ± 24 |

**Table 4** Concentrations of non-volatile impurities in $UF_6$ samples

Tables 3 and 4 show the samples have a total impurity concentration lower than 5.0 mg in 100 g of sample, which results in a purity level of 99.995 %. In this way the material can be considered sufficiently pure to be used as isotopic reference materials in the most demanding measurement processes.

## 4.2 Isotopic characterization results

The measurement results for the isotope amount ratio $n(^{235}U)/n(^{238}U)$ are presented in table 5. The absolute values of expanded uncertainty ($U$), calculated with coverage factor ($k$) equal to 2, are presented between parentheses after the isotope ratios. Their relative values (%) are presented in a separated column.

| MRI | $n(^{235}U)/n(^{238}U)$ | $U$ ($k$=2) |
|---|---|---|
| | | % |
| **0.5** | 0.005 354 7 (17) | 0.032 |
| **0.7** | 0.007 254 3 (16) | 0.022 |
| **1.0** | 0.010 370 3 (18) | 0.017 |
| **2.5** | 0.024 232 0 (42) | 0.017 |
| **3.5** | 0.035 469 8 (47) | 0.013 |
| **4.5** | 0.046 545 7 (65) | 0.014 |
| **6.5** | 0.069 850 (23) | 0.033 |
| **10** | 0.107 545 (90) | 0.084 |
| **15** | 0.182 38 (18) | 0.10 |
| **20** | 0.254 42 (28) | 0.11 |

**Table 5** Results of isotope amount ratio $n(^{235}U)/n(^{238}U)$ and expanded uncertainties ($U$) obtained by GSMS technique

Table 5 shows that the relative expanded uncertainty values for all samples are lower than 0.11 % and seven samples of the set have uncertainties even lower than 0.05 %. This is remarkable achievement because the expanded uncertainty estimated according to the ISO-GUM is a quantification of the reliability of the measurement performed.

The uncertainties of samples MRI 10, 15 and 20 are three to four times higher than the others in the set. This is due to the need to use isotope reference materials for these three samples with higher expanded uncertainties than those used for the remaining samples.

The analysis of the uncertainty budget of each sample revealed that the standard uncertainty is composed by two main components: the uncertainty of the isotope reference materials used and the repeatability of measurements required by the analytical method employed.

As the dominant component in the budget is the first one, the increase in the uncertainty of the reference materials used in the measurement process implies an increase in the uncertainties of the three mentioned samples.

The relative contributions of these two components to the standard uncertainty ($k$=1) of the isotope ratio of sample MRI 2.5 is presented in figure 2.

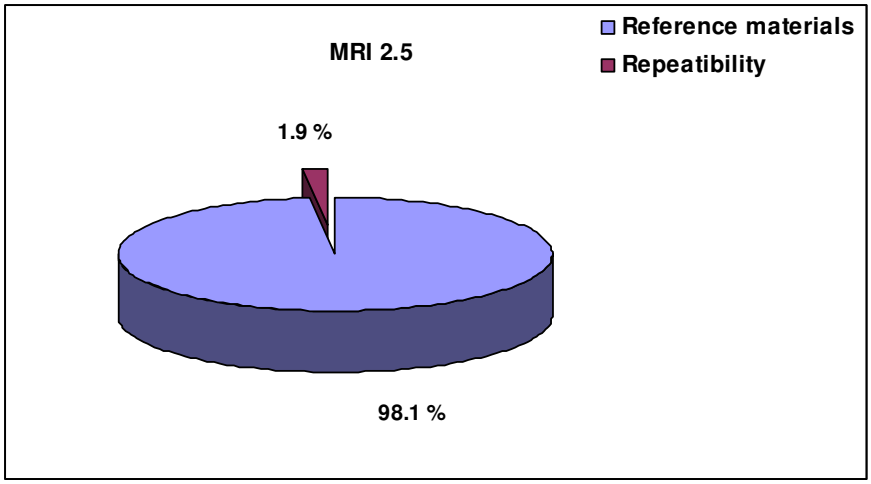



**Figure 2** Relative contribution of the components to the standard uncertainty of the isotope amount ratio $n(^{235}U)/n(^{238}U)$ for sample MRI 2.5 measured by GSMS

The measurement results for the isotope ratios $n(^{234}U)/n(^{238}U)$ and $n(^{236}U)/n(^{238}U)$ are presented in table 6.

| MRI | $n(^{234}U)/n(^{238}U)$ | $U$ % | $n(^{236}U)/n(^{238}U)$ | $U$ % |
|---|---|---|---|---|
| **0.5** | $3.581\ 2\cdot10^{-5}$ (45) | 0.13 | $1.148\ 0\cdot10^{-6}$ (32) | 0.28 |
| **0.7** | $5.658\ 1\cdot10^{-5}$ (41) | 0.07 | $3.213\ 3\cdot10^{-8}$ (89) | 0.28 |
| **1.0** | $8.70\cdot10^{-5}$ (10) | 1.15 | $2.917\cdot10^{-6}$ (40) | 1.37 |
| **2.5** | $2.092\ 8\cdot10^{-4}$ (13) | 0.06 | $1.140\ 8\cdot10^{-7}$ (31) | 0.27 |
| **3.5** | $3.327\ 1\cdot10^{-4}$ (18) | 0.05 | $3.881\ 0\cdot10^{-4}$ (11) | 0.03 |
| **4.5** | $4.411\cdot10^{-4}$ (98) | 2.22 | $5.44\cdot10^{-4}$ (12) | 2.21 |
| **6.5** | $8.725\cdot10^{-4}$ (78) | 0.90 | $1.697\cdot10^{-4}$ (20) | 1.18 |
| **10** | $1.032\ 2\cdot10^{-3}$ (67) | 0.65 | $9.90\cdot10^{-4}$ (14) | 1.40 |
| **15** | $1.744\ 4\cdot10^{-3}$ (44) | 0.25 | $1.695\ 5\cdot10^{-3}$ (59) | 0.35 |
| **20** | $2.404\ 4\cdot10^{-3}$ (12) | 0.05 | $2.323\ 39\cdot10^{-3}$ (59) | 0.03 |

**Table 6** Results of the isotope amount ratios $n(^{234}U)/n(^{238}U)$ and $n(^{236}U)/n(^{238}U)$ and expanded uncertainties obtained by TIMS technique

Although all samples were initially measured using a Finnigan MAT 262 mass spectrometer, the most representative and therefore important samples of the set (MRI 0.5, 0.7, 2.5, 3.5 and 20) were re-measured using a Finnigan Triton, a mass spectrometer having more powerful measuring resources.

The measurement results from both instrumentations were in close agreement and must be considered statistically equivalent if their expanded uncertainties are taken into account. Yet the results provided by the Triton always presented lower uncertainty values as can be seen in table 6.

The analysis of the uncertainty budget for isotope amount ratio $n(^{234}U)/n(^{238}U)$ revealed that the isotope reference materials used and the repeatability have varying contributions depending on the $^{234}U$ concentration in the sample.

The relative contributions obtained for sample MRI 2.5 are presented in figure 3 as an example

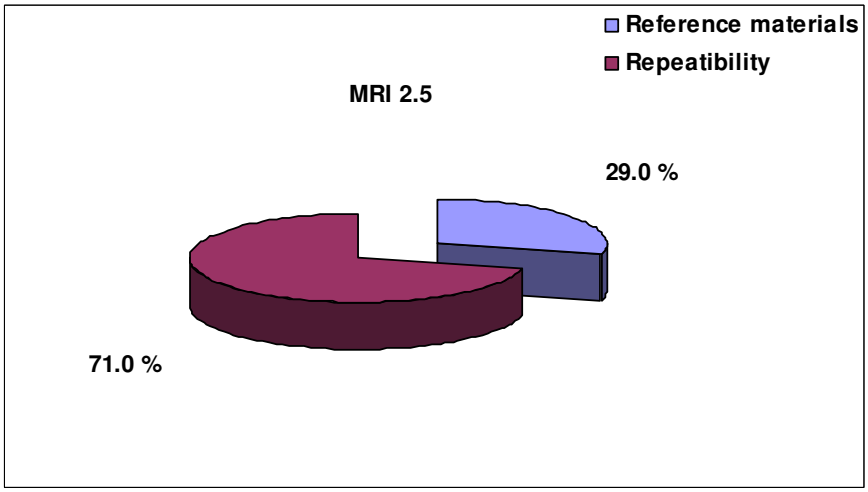



**Figure 3** Relative contribution of the components to the standard uncertainty of the isotope amount ratio $n(^{234}U)/n(^{238}U)$ for sample MRI 2.5 measured by TIMS-Triton

Analysis of the uncertainty budget of the isotope amount ratio $n(^{236}U)/n(^{238}U)$, presented in figure 4, revealed that a significant uncertainty is introduced from the need to calibrate the SEM device. Should this component be neglected, the value of the total uncertainty would be seriously underestimated.

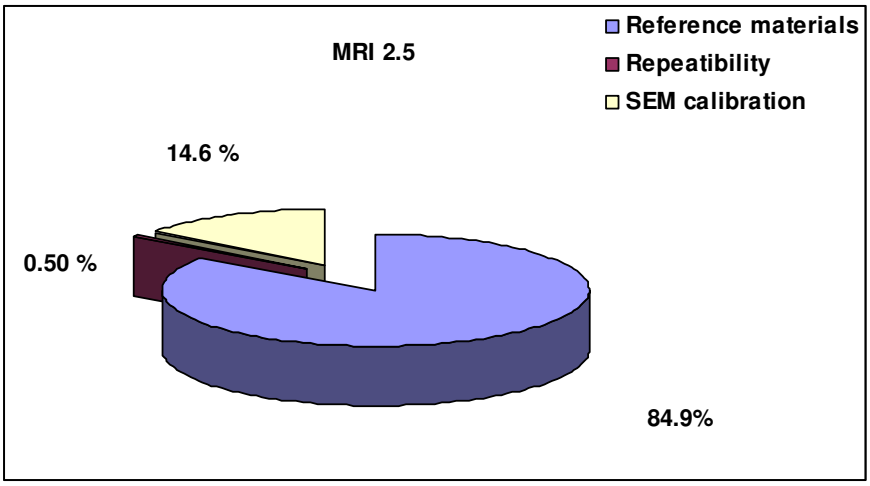



**Figure 4** Relative contribution of the components to the standard uncertainty of the isotope amount ratio $n(^{236}U)/n(^{238}U)$ for sample MRI 2.5 measured by TIMS-Triton

## 4.3 Sample stability, homogeneity and storage

Material stability was not a matter of concern in this programme because $UF_6$ is stable up to 1 000 K.

However, the compound it is extremely reactive, especially when in contact with water. It must therefore be stored in stainless steel ampoules, assembled with high vacuum valves and helium leak tested.

At the end of the experimental part of this programme, the $UF_6$ samples stored in ampoules 1S were homogenised for 1 h at 90 C and then transferred to ampoules 3S. A final measurement confirmed the thorough isotopic homogeneity in each ampoule.

The ampoules 1S (larger one) and 3S (smaller one) containing the isotope reference material MRI 2.5 can be seen in figure 5.


**Figure 5 Ampoules 1S and 3S used to store isotope reference materials MRI 2.5**

## 4.4 Certification of the isotope reference materials produced

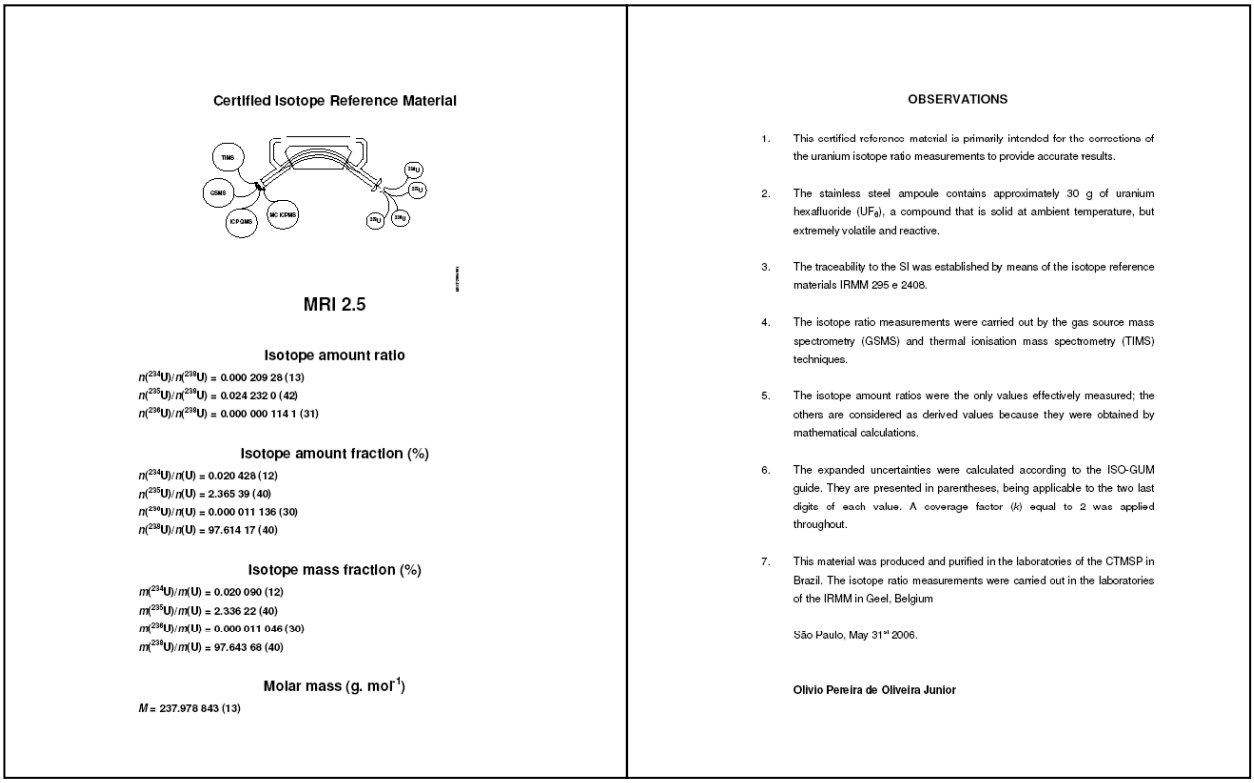
The certificate associated to each material produced brings its official code number, the measured isotope amount ratio and other very important analytical data like its isotope amount fraction, isotope mass fraction and molar mass.

These last values are considered derived values because they were calculated from the measured isotope amount ratios presented in tables 5 and 6.

The intended use of the material, the amount of $UF_6$ within the ampoule, the particular traceability link to the SI and other valuable information are also written in the certificate.

The definition of an expiry date was deemed not applicable for this kind of material.

The official certificate of the Isotope Reference Material MRI 2.5 is presented in figure 6 as an example.



Certified Isotope Reference Material

MRI 2.5

Isotope amount ratio

$n(^{234}U)/n(^{238}U)$ = 0.000 209 28 (13)
$n(^{235}U)/n(^{238}U)$ = 0.024 232 0 (42)
$n(^{236}U)/n(^{238}U)$ = 0.000 000 114 1 (31)

Isotope amount fraction (%)

$n(^{234}U)/n(U)$ = 0.020 428 (12)
$n(^{235}U)/n(U)$ = 2.365 39 (40)
$n(^{236}U)/n(U)$ = 0.000 011 136 (30)
$n(^{238}U)/n(U)$ = 97.614 17 (40)

Isotope mass fraction (%)

$m(^{234}U)/m(U)$ = 0.020 090 (12)
$m(^{235}U)/m(U)$ = 2.336 22 (40)
$m(^{236}U)/m(U)$ = 0.000 011 046 (30)
$m(^{238}U)/m(U)$ = 97.643 68 (40)

Molar mass (g. mol$^{-1}$)

$M$ = 237.978 843 (13)

OBSERVATIONS

1. This certified reference material is primarily intended for the corrections of the uranium isotope ratio measurements to provide accurate results.
2. The stainless steel ampoule contains approximately 30 g of uranium hexafluoride ($UF_6$), a compound that is solid at ambient temperature, but extremely volatile and reactive.
3. The traceability to the SI was established by means of the isotope reference materials IRMM 295 e 2408.
4. The isotope ratio measurements were carried out by the gas source mass spectrometry (GSMS) and thermal ionisation mass spectrometry (TIMS) techniques.
5. The isotope amount ratios were the only values effectively measured; the others are considered as derived values because they were obtained by mathematical calculations.
6. The expanded uncertainties were calculated according to the ISO-GUM guide. They are presented in parentheses, being applicable to the two last digits of each value. A coverage factor ($k$) equal to 2 was applied throughout.
7. This material was produced and purified in the laboratories of the CTMSP in Brazil. The isotope ratio measurements were carried out in the laboratories of the IRMM in Geel, Belgium

São Paulo, May 31st 2006.

**Olivio Pereira de Oliveira Junior**

**Figure 6** Certificate of the Isotope Reference Material MRI 2.5

## 5. Conclusions

A set of ten $UF_6$ isotope reference materials in the range of 0.5 to 20.0 % $^{235}U$ in mass were successfully prepared, purified, characterized and certified.

These samples are traceable to the SI because isotope reference materials ultimately linked to synthetic mixture of highly enriched oxides were employed in the measurement process.

The measurement uncertainties of the isotope ratios $n(^{234}U)/n(^{238}U)$, $n(^{235}U)/n(^{238}U)$ and $n(^{236}U)/n(^{238}U)$ are in the range of de 0.05 to 2.22 %, 0.013 to 0.10 % and 0.03 to 2.21 % respectively.

This set of materials will be used not just in the measurement process of ordinary samples, but also in the calibration of mass spectrometers, quality control routines and evaluation of the performance of new instruments, methods and techniques.

The utilization of these materials will eventually allow the achievement of isotope amount ratios associated with much lower uncertainty values for the safeguards samples processed in Brazilian laboratories.

Consequently it will contribute to the improvement of the reliability and trust of the nuclear material safeguards system in South America.

**This paper is dedicated to Dr. Roger Wellum on the occasion of his retirement**

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# Session 6

# Proliferation Resistance - II

# Proliferation Resistance Characteristics of Advanced Nuclear Energy Systems: a Safeguardability point of view

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***Abstract:***

*Among the international community there is a renewed interest in nuclear power systems as a major source for energy production in the near to mid future. This is mainly due to concerns connected with future availability of conventional energy resources, and with the environmental impact of fossil fuels. International initiatives as the Generation IV International Forum (GIF), the International Project on Innovative Nuclear Reactors and Fuel Cycles (IAEA-INPRO), partially the US driven Global Nuclear Energy Partnership (GNEP), aimed at defining and evaluating the characteristics in which future innovative nuclear energy systems (INS) will have to excel, have been set up. Among the identified characteristics, Proliferation Resistance plays an important role for being able to widely deploy nuclear technology worldwide in a secure way.*

*Studies having the objective to assess Proliferation Resistance of nuclear fuel cycles have been carried out since the seventies, e.g. the International Nuclear Fuel Cycle Evaluation (INFCE) and the Nonproliferation Alternative Systems Assessment Program (NASAP) initiatives, and all agree in stating that absolute intrinsic proliferation resistance, although desirable, is not achievable in the foreseeable future. The above finding is still valid; as a consequence, every INS will have to comply with Non Proliferation Treaty (NPT) connected agreements and will require safeguards measures, implemented through extrinsic measures. This consideration led to a renewed interest in the "Safeguardability" concept which can be seen a as a bridge between intrinsic features and extrinsic features and measures.*

**Keywords:** Proliferation Resistance, Safeguardability, Innovative Nuclear Energy Systems, Holistic Approach, Evaluation/Assessment.

## 1. Introduction

While not a completely new concept, Safeguardability has often been connected to nuclear material characteristics. The GIF Proliferation Resistance & Physical Protection Working Group (GIF PR&PP WG) recognised that this is a reductive view of the subject, and preliminary began to address it in a more comprehensive way. PR&PP WG has more broadly defined safeguardability as the degree of ease with which a system can be effectively and efficiently put under international Safeguards.

Although not required for the PR&PP evaluation framework, the safeguardability concept has been recommended for use in the PR&PP approach because it can be of support to designers to consider safeguards needs beginning with the earliest phases of design. A preliminary list of attributes affecting the safeguardability of an Innovative Nuclear System, including attributes that affect material accounting, containment and surveillance, and design information verification, has been identified in the GIF PR&PP methodology report revision 3 and has been already reported [1, 2]. The current list of the safeguardability attributes is reported in Appx. D of the Revision 5 of the PR&PP methodology report [3, 4]. JRC is among the promoters of Safeguardability in GIF PR&PP EG, and starting from the

[1] Also enrolled as a PhD student at University of Bristol under the supervision of Prof. D. Blockley.

work that is being carried out within the group, is interested in addressing the Safeguardability concept from a holistic point of view.

In section 2 the paper will firstly recall how the issue of Safeguardability had been approached in the past and then it will give a summary of the Safeguardability concept so far emerged within the PR&PP expert group activity (section 3). In section 4 a possible way forward for the Safeguardability concept in the GIF PR&PP frame is proposed. In a second part of the paper, and section 5 highlights a possible redefinition of some key concepts in a holistic approach to Safeguardability. Finally, in section 6 some conclusions are presented.

## 2. Safeguardability in Literature

The issue of how to tackle the problem of the Safeguardability of nuclear energy systems existed since the foundation of the International Atomic Energy Agency, and during the last thirty years at least two different approaches emerged:

- Developing new safeguarding techniques and equipments to enhance safeguards effectiveness and efficiency.
- Providing guidelines for designers of new systems in order to enhance systems Safeguardability during early design stages.

In the following paragraphs these two different approaches will be briefly described.

### 2.1. Developing new safeguarding techniques and equipments to enhance safeguards effectiveness and efficiency

Although nuclear proliferation concerns have always accompanied the development of civilian nuclear technologies, the standard approach to non proliferation didn't use to consider Safeguardability as part of the design requirements of nuclear energy system. Typically a nuclear energy system was firstly designed and licensed, and in a following phase the IAEA and the State in the process of building the system negotiated the safeguards approach for it. This implied that safeguarding techniques and equipments were conceived and/or adapted for the nuclear energy system design after the design was already fixed, with very limited capabilities of changing those design aspects that might create Safeguardability issues. The main R&D activities connected to safeguards were therefore those aimed at enhancing the effectiveness and efficiency of detection equipments and inspection planning (see e.g. [5,6]), and the few assessments in the field of safeguards had the objective of evaluating the effectiveness of a given safeguards approach applied to a given system (an overview of some of such studies can be found in [7]).

This approach has been favoured by the fact that up to the Nineties systems haven't changed much from a safeguarding point of view, and therefore efforts have been put in advancing in the technologies connected to detection equipments. Recently, with the start-up of international initiatives whose objective is to design the next generation of nuclear energy systems, the safeguarding community faces a new scenario, involving the need of new safeguards techniques able to cope with the major changes in the nuclear processes considered for new systems. Examples of discussions about the Safeguardability of innovative fuel cycles in relation to the current safeguarding practice and capability can be found in [8] and [9], where the Safeguardability of a pyroprocessing facility [8] and of an advanced spent fuel conditioning process [9] are discussed, or in [10], where the Safeguardability of an innovative closed fuel cycle is under examination.

In [8], the Los Alamos National Laboratory carried out an analysis of the Safeguardability of a pyroprocessing facility under various assumptions of both design and safeguards approaches [11]. This exercise is particularly interesting because it tackles the issue of Safeguardability form a double point of view: not only by varying safeguards measures and techniques, but also by varying the original process design in order to make it more safeguardable.

In [9], the Safeguardability of an advanced spent fuel conditioning process developed in Korea is analysed by a study jointly carried out by the Los Alamos National Laboratory and the Korean Atomic Energy Research Institute (KAERI). After giving some hints of the innovative characteristics of the

process leading to the absence of separated Pu (i.e. discussing material attractiveness, an aspect related to proliferation resistance), a hypothetical safeguards approach is assumed in order to highlight whether IAEA detection goals are achievable or not.

In [10], EC DG-JRC ITU analysed a hypothetical closed fuel cycle (double strata) from a double point of view: a proliferation resistance point of view and a Safeguardability one. Proliferation resistance is mainly analysed from a material quality point of view, following the concept that proliferation resistance of a nuclear system may be increased by reducing the attractiveness of the involved nuclear material [12]. Safeguardability is then analysed by reasoning on the possibility to set up effective safeguards measures with current detection equipments and analytical techniques, and eventually hints are provided on the R&D activities to be carried out to cope with actual limitations. It is interesting to note how the two aspects of proliferation resistance and of Safeguardability are analysed separately and with different criteria. The relationship between these two aspects is briefly discussed in the next section of this paper.

These studies put in evidence how assessing Safeguardability of systems whose designs are not yet finalised can open a double front: on one side there's the possibility to understand how current safeguarding techniques and practices are suited for new concepts, and on the other side there's the possibility to propose design changes to enhance systems Safeguardability.

## 2.2. Providing guidelines for designers of new systems in order to enhance systems Safeguardability during early design stages

Currently, national and international initiatives aimed at developing the next generation of nuclear energy systems have been launched. Among them, the two major international initiatives are the *Generation IV International Forum* [13] and the IAEA *International Project on Innovative Nuclear Reactors and Fuel Cycles* (INPRO) [14] initiatives. Both efforts set up a number of goals that future systems will have to reach, and among them there's a call for increased proliferation resistance.

Proliferation resistance has been defined by [15] as:

*That characteristic of an NES that impedes the diversion or undeclared production of nuclear material or misuse of technology by the Host State seeking to acquire nuclear weapons or other nuclear explosive devices.*

*The degree of proliferation resistance results from a combination of, inter alia, technical design features, operational modalities, institutional arrangements and safeguards measures.*

Proliferation resistance is therefore considered to be made of *intrinsic* features and *institutional* measures concurring in making the nuclear energy system unattractive for its use in a military programme. Innovative nuclear energy systems will therefore still rely on international nuclear safeguards, and this should be considered a design requirement. It has to be noted that even if the INPRO initiative doesn't talk openly about Safeguardability, this concept is *de facto* present behind indicators they identified as relevant to detectability.

Although no one is entitled to impose proper design requirements related to systems Safeguardability, systems designers repeatedly asked the PR&PP WG for guidelines able to help them to take this aspect into account in their activity. Scanning past literature for studies providing guidelines for designing more safeguardable nuclear energy systems highlights that this topic has been investigated since the late Seventies, but relatively few studies have been carried out, mainly in the US [16] and by IAEA [17, 18]. Their main focus is on nuclear reactors, and they provide a good basis for preparing guidelines for designing future systems.

## 3. Safeguardability within the GIF PR&PP Evaluation Methodology

Although not required for the PR&PP evaluation framework, the safeguardability concept has been recommended for use in the PR&PP WG because it can be of support to designers to consider safeguards needs beginning with the earliest phases of design, and is currently described in Appendix D [4] of the Rev.5 of the PR&PP Evaluation Methodology report [3].

### 3.1 Safeguardabilty as in GIF PR&PP Evaluation Methodology report, Rev. 5, app. D

In [4], Safeguardability is defined as *the degree of ease with which a system can be effectively and efficiently put under international Safeguards.* Its analysis is strongly influenced by a number of aspects connected with a system's design, such as system's layout, the type of process chosen and its actual implementation, the foreseen operating profile. In addition, another relevant source of influence is the international non proliferation legal framework under which the system will be operated.

From its definition, Safeguardability will be related to the potential easiness of setting up a safeguards approach which would effectively and efficiently provide credible assurance that no undeclared illicit activities have been carried out in the nuclear energy system. This implies that a Safeguardability analysis is based on the knowledge of the nuclear energy system, of the legal framework in which the nuclear energy system is operated and on the verification activities and techniques at disposal within the considered legal framework. Any change not only to the first aspect, but also to the other two will lead to a different result of the Safeguardability analysis, making it evolutionary in nature and context dependent.

Two possible objectives were considered for Safeguardability:

The first one was to have a possible substitute for two pathways measures - Detection Probability and Detection Resources Efficiency - where not enough information for their estimation is available. Typically this is the case of very early design stages.

The second one was to answer to a systems designers request to have some kind of guidance for designing systems able to ease activities connected with the implementation of international safeguards.

For analysing Safeguardability, a set of attributes have been identified by the Expert Group, grouped in three broad categories: 1) attributes capturing the potential ease of performing Design Information Verification (DIV), 2) attributes capturing the potential ease of performing Nuclear Material Accounting (NMA) and finally 3) attributes capturing the potential ease of implementing Containment and Surveillance (C/S). For each category, a table has been prepared, containing the relevant attributes and a short description of their meaning.

Attributes have been identified as the system's intrinsic features upon which current safeguards techniques rely, independently of the fact that such technique will be actually implemented. In analysing Safeguardability, no assumption of a particular safeguards approach is made. [4] states that *whereas Safeguardability is an index of the potential ease of implementing an effective and efficient Safeguards approach based on current practice, Detection Probability is an index of the effectiveness of an implemented Safeguards approach.*

The current tables are published in [4], and represent a first step towards the identification of the systems' design attributes that may affect Safeguardability. The tables there reported have been developed considering two different scenarios:

**In the first scenario** a team of experts is to perform a Safeguardability assessment on the system's design in order to provide relevant feedbacks to the system's design team. Designers would hence have the possibility to improve the design according to the recommendations emerged from the assessment. This close and iterated interaction between designers and Safeguardability experts would eventually lead to a highly safeguardable nuclear energy system.

**In the second scenario**, The Safeguardability assessment is performed for supporting policy makers in decisions where a high Safeguardability is an important characteristic.

This double approach led to a set of attributes that are general in nature, partly characterised having a feedback to designers in mind, and partly characterised having a feedback to policy makers.

In the following paragraph, the first scenario has been selected for illustrating a possible way forward for advancing in the work.

## 3.2. Advancing in the work: an improved characterisation of the identified attributes

The first step for progressing is to validate the current tables and investigating their completeness. These tables have undergone a validation exercise within the PR&PP WG. Since they are not to be considered as finalised but only a first milestone in a work in progress, the tables are currently undergoing an external validation process via interviewing relevant domain experts not directly involved in the GIF PR&PP activities. Both the internal and the ongoing "external" review process put in evidence that the first step in the scheduled advancement in the work is to better characterise the identified attributes. Ideally, each attribute could be characterised according to six aspects, identified by the six keywords *What*, *Why*, *Who*, *How*, *When*, and *Where* (this paradigm is inspired by and adapted from [19]). For each attribute, the objective is to fill Table 1, adapted from [20], where the What aspect captures the objective of the attribute, the Why captures the rationale, and eventually some examples where this attribute is fulfilled and not fulfilled, the Who captures all the players for whom the attribute is relevant and the experts needed for assessing it, The How captures the performance indicators that could be used for assessing the attribute and its related description and scale, the When and Where are grouped for capturing the stage at which the attribute comes into the game.

| | | | | |
|---|---|---|---|---|
| **Name** | | | | |
| **What** | | | | |
| **Why** | *Rationale:*<br>*Example(s):* | | | |
| **Who** | *Relevant to:*<br>*Assessed by:* | | | |
| **How** | *Performance Indicator* | *Description* | *Scale* | *Comment* |
| **When/ Where** | | | | |

**Table 1:** Attributes Characterisation table in terms of What, Why, Who, How, When/Where. Adapted from [20].

In this paper an attempt to characterise the Safeguardability attributes identified in [4] in terms of Objective (What) and Rationale (Why) is presented in Tables 2, 3 and 4. These tables build on the official ones in [4], and although a brainstorming for identifying new missing attributes to be added to the current ones is in course, this effort has not been included here. As a consequence, no additional attributes are here introduced, and the focus is uniquely on advancing with the available material.

A Safeguardability analysis intended at providing feedback at systems designer teams should be capable of analysing the systems design at the earliest possible design stage. Clearly not all the attributes identified in Tables 2-4 can be relevant at very early design stages, some of them might require a fairly detailed description of the system. Since every attribute is considered to be important for a sound Safeguardability analysis, this aspect calls for a method able to deal with incompleteness of information.

| ***Name*** | ***Objective (What) and Rationale (Why)*** |
|---|---|
| Comprehensiveness of facility documentation and data. | **Objective**: Making sure that the facility documentation is exact and complete in all the aspects relevant to design verification activities.<br><br>**Rationale**: Every facility that has to be put under international safeguards will have to be described in a documentation set requested by the safeguards inspectorate. For Design Information Examination (DIE) and DIV, a Design Information Questionnaire (DIQ) has to be compiled. Exact and complete documentation about the facility, with detailed layout in both hardcopy and electronic form would greatly facilitate a comprehensive compilation of DIQ, and this will in turn benefit the inspectorate for the foreseen activities. |
| Transparency of layout | **Objective**: To make sure that the system layout is conceived in such a way that process lines are easily identifiable and could be checked for consistency with the declarations at different design stages<br><br>**Rationale**: An important aspect of Design Information Verification (DIV) is verification of the process equipments and layout. Often process equipments layout is not conceived for easy layout verification, and is therefore difficult to check. Although this might not be an issue in item facilities, it might create big difficulties when bulk facilities are involved, especially when the process is continuous. |
| Possibility to use computerised reconstruction models | **Objective**: To make sure that the system is conceived in such a way that technologies aimed at making the DIV activity easier could be used.<br><br>**Rationale**: Modern techniques such as the ones based on 3D laser rangefinders allow performing a 3D mapping of the area to be verified, thus, allowing computer assisted verification of the equipments and facility layout. |
| Possibility to have visual/instrumented access to facility equipments while operational. | **Objective**: To make sure that the system is conceived in such a way that every relevant process equipment can be visually or instrumentally checked for DIV purposes during the facility normal activity.<br><br>**Rationale**: DIV are normally performed during facilities' planned shut downs, but this is not always possible. In addition, on some facilities, accessibility to all relevant equipments is not possible even during planned shut downs, due to e.g. radiological hazards. Compliance with this attribute would ease the work of the inspectors and avoid any loss of time/resources to the operator. |

**Table 2:** Nuclear System Attributes facilitating Design Inventory Verification (DIV).

| ***Name*** | ***Objective (What) and Rationale (Why)*** |
|---|---|
| Uniqueness of material signature | **Objective**: Making sure that the nuclear material available in the facility has intrinsic characteristics that contribute to easily recognise it in terms of type and composition.<br><br>**Rationale**: Having unambiguous material signature positively affects measures aimed at discriminating the nuclear material composition. Moreover it makes any concealment activity aimed at substituting the declared nuclear material with dummies more difficult.<br><br>In origin inspired by [21] |
| Hardness of material signature. | **Objective**: Making sure that the nuclear material available in the facility has a radiation signature which is easy to be measured.<br><br>**Rationale**: During PIV, inspectors typically perform attribute verification measurements on a sample of the available nuclear material. Having hard nuclear material signature would facilitate the inspectorate to detect and record it and therefore conclude that the measured material is compliant with the operator's declarations.<br><br>In origin inspired by [21] |
| Possibility of applying passive measurement methods | **Objective**: Making sure that the nuclear material available in the facility can be characterized via passive measurement methods instead of requiring active measurement methods.<br><br>**Rationale**: During PIV inspectors typically perform attribute verification measurements on a sample of the available nuclear material. If the nuclear material characteristics allow passive NDA techniques to be used to reach the intended objective, the inspectorate would have the opportunity to get the job done in an easier and cheaper way.<br><br>In origin inspired by [21] |

| ***Name*** | ***Objective (What) and Rationale (Why)*** |
|---|---|
| Item/bulk | **Objective**: Making sure that the nuclear material form inside the system is compatible with accurate and efficient NMA.<br><br>**Rationale**: In terms of nuclear material accounting, item facilities are easier and cheaper to verify than bulk facilities, since Material Balance is easier to be closed (in principle no MUF is expected) and verifications typically rely on NDA measurements rather than DA measurements. In principle NMA in items facilities is less resources intensive than bulk facilities.<br><br>In origin inspired by [21] |
| Uncertainties of detection equipments | **Objective**: Making sure that the system is designed in such a way that the nuclear material treated can be verified and inventoried with current verification techniques and equipments.<br><br>**Rationale**: New systems might take advantage of new processes and possibly innovative nuclear material forms. Designing the system keeping in mind that those processes will have to cope with NMA, carried out using specific verification techniques, with know performance target values, means to facilitate the safeguards designers work and reduce the safeguards resources needed to be allocated.<br><br>In origin inspired by [21] |
| Annual throughput | **Objective**: Making sure that the amount of nuclear material produced by the system's processes is compatible with the international safeguards inspection goals.<br><br>**Rationale**: Large bulk facilities handling big quantities of nuclear material per year might generate throughputs challenging the possibility to reach the safeguards detection goals. Annual throughput might affect the accuracy of physical inventories in absolute terms, and therefore detection limits. |
| Batch/continuous process | **Objective**: Making sure that the type of material handling, within the process chosen for the facility, is compatible with an accurate closing of a nuclear material balance.<br><br>**Rationale**: How nuclear material is handled in a bulk facility might have a strong impact on the overall difficulty of closing the material balance. In principle batch processes, possibly with constant nuclear material composition between batches, could lead to easier and more accurate closing of material balances when compared with continuous processes. |
| Radiation Field | **Objective**: Making sure that the system is designed in such a way that the nuclear material radiation field doesn't affect the inspection activities.<br><br>**Rationale**: The presence of a radiation field generated by the system's processes and involved nuclear material is unavoidable. A safeguardable facility should be designed in such a way that the inspectorate's activities are not jeopardised by radiological hazards. This affects both the inspection activities and the eventual servicing of fixed equipment (accessibility etc.). |
| Amount of hidden (unverifiable) inventory | **Objective**: Making sure that the system's design is optimised for minimizing the amount of nuclear material that might not be accessible to inspectors during safeguarding activities<br><br>**Rationale**: Each process has a physiological amount of nuclear material that is not accessible by inspection activities because e.g. inside process pipelines. A system optimised to minimise this amount of material would facilitate the closing of the material balance by the inspectorate. |
| Possibility to implement near real time accountancy | **Objective**: Making sure that the system's design is compatible with the implementation of near real-time accounting techniques.<br><br>**Rationale**: For some processes the closure of material inventory for timeliness purposes is particularly challenging. Having the possibility to implement near real time accounting would greatly help the inspectorate in closing the material balance frequently and without interrupting the process, in order to achieve timeliness objectives while not intruding in the system's operation. |

**Table 3:** Nuclear System Attributes facilitating Nuclear Material Accounting (NMA).

| ***Name*** | ***Objective (What) and Rationale (Why)*** |
|---|---|
| Operational practice | **Objective**: Making sure that the systems operational profile and procedures facilitate the applicability of C/S measures.<br><br>**Rationale**: The aim of C/S measures is to maintain continuity of knowledge on the systems' nuclear material inventory between two inventory verifications. The way in which the system is operated might increase the ease of applying containment and surveillance measures. |
| Extent of automation and remote handling | **Objective**: Making sure that procedures are carried out with a low need of human intervention and that all operator's equipments can be instrumented by the inspectorate.<br><br>**Rationale**: Having highly automated processes would facilitate the application of C/S measures in several ways: collection of data coming from operation equipments can be used for continuity of knowledge purposes, and the possibility to have few personnel in the operations area would facilitate the review of surveillance cameras recordings. |

| ***Name*** | ***Objective (What) and Rationale (Why)*** |
|---|---|
| Standardisation of items in transfer | **Objective**: Making sure that items in transfer inside the system are as standardised as possible.<br><br>**Rationale**: Having standardised items (e.g. flasks) in transfer would facilitate the application of C/S measures in several ways. Examples are: easier interpretation of recorded images in the review phase, easier surveillance of standard items in transit as they would probably result in moving at the same speed. |
| Possibility to apply optical surveillance | **Objective**: Making sure that the system's design is optimised for the use of optical surveillance devices.<br><br>**Rationale**: In current practice surveillance relies heavily on the use of images recorded by surveillance cameras. In order to maximise the benefits that this technique embeds, the system can be designed keeping this in mind. |
| Number of possible transfer routes for items in transit | **Objective**: Making sure that the system's design foresees a limited amount of possible transfer routes for the nuclear material in transit.<br><br>**Rationale**: Having only one or very limited possible transfer routes for items in transit would greatly improve easiness of performing surveillance and interpreting surveillance records during the review phase.<br>This attribute is also particularly important for facilitating the application of containment measures, given that operational rules allow their use (e.g. no transfers for long period of times). |
| Possibility to apply remote surveillance | **Objective**: Making sure that the system's design allows the possibility to transfer C/S data offsite.<br><br>**Rationale**: Remote surveillance helps to achieve timeliness and saves onsite inspection efforts, concurring in lowering the resources needed by a safeguards approach. |

**Table 4:** Nuclear System Attributes facilitating application of Containment and Surveillance (C/S) and other monitoring systems.

## 4. The way forward within the PR&PP WG framework

In addition to characterising the current attributes by filling Table 1 for each of them, the ongoing review of the current Safeguardability tables put in evidence a number of aspects that are going to be deepened and implemented in the next steps. Among the aspects that will be deepened in the near future are the following:

- The current tables are mainly focused on the activities to be carried out during routine inspections. It is foreseen to give more coverage to those aspects that are relevant for providing the Safeguards Inspectorate with the information needed for designing the system's safeguards approach, e.g. the information needed for compiling the system's *Design Inventory Questionnaire* (DIQ) and the facility attachments.
- No particular attention has been devolved yet to the aspects influencing the easiness of collecting the information needed to the operator for the reporting activities foreseen by the Inspectorate. It is desirable to take explicitly into account those aspects facilitating the setting up of an accurate and efficient accounting and reporting systems by the operator.
- Aspects critical for the Safeguardability of a nuclear energy system but not considered yet are those that influence the easiness of e.g. recovering from interruption of continuity of knowledge, or of performing typical follow-up actions. It is foreseen to enhance the coverage of these aspects.
- During the preliminary investigation of the Safeguardability concept, the PR&PP WG deliberately focused only on the traditional Safeguards measures and activities. It is foreseen to begin the investigation of the aspects related to the measures introduced by the Additional Protocol and the subsequent Strengthened Safeguards regime.

Once the attributes are finalised, there's the need of a mean to capture the evidence for and/or against their fulfilment, in order to be able draw a conclusion on every single attribute and eventually, on the basis of the conclusions on the single attributes, to be able to draw a conclusion on the overall Safeguardability of the system under investigation. Any candidate technique for this task will have as minimum requirement to be able to cope with conflicting and incomplete evidence, and to be able to capture the involved uncertainty in all its forms, i.e. fuzziness, incompleteness and randomness.

# 5. Some reflections on the Safeguardability concept: setting up the case for a holistic approach

## 5.1. Safeguardability analysis on what? Setting the scale

Since the Safeguardability concept is aimed at providing feedback to systems designers, it is worthwhile to spend some time investigating the meaning of nuclear energy system. The actual definition of nuclear energy system within the GIF PR&PP Evaluation Methodology is the following:

*A Generation IV Nuclear Power Producing Plant and the facilities necessary to implement its related fuel cycle*[2].

Actually this definition implies that the nuclear energy system is the whole nuclear fuel cycle involving an innovative nuclear power reactor. It is very unlikely that a single design team will address the whole nuclear fuel cycle, and therefore it would be reasonable to allow the Safeguardability analysis to be performed also on a single facility or even to a single process of a particular facility. In principle, the scope of the analysis will define if the analysis will be performed on a process line, on a facility or on a complete fuel cycle. If this is accepted, it would be possible to notice how the Safeguardability concept could be defined in a holistic way, and suited to all needed scale.

The term *holistic* is connected with the concept of *holon*: a holon is at the same time a part and a whole. For example the human being is a whole for it is made of different sub-systems as the skeleton, the cardiovascular system, the pulmonary system, etc. At the same time it is a part since it is part of a social structure (family, city, nation, etc.)[3]. Whether a human being is considered to be a part or a whole is a matter of the scope of the analysis and of the chosen level of detail. The same thing applies to our problem: depending on the scope, a facility can be considered as a part of a system or of a fuel cycle or as a whole for the processes carried out in it. In principle a Safeguardability analysis could be performed regardless of the chosen holon. It has to be noticed that passing from e.g. a facility scale to a fuel cycle scale, the Safeguardability of the latter generally *emerges* from the integrated behaviour of the former ones, i.e. the Safeguardability of the fuel cycle might be different from a simple aggregation of the Safeguardability of its single facilities.

## 5.2. Safeguardability: a concept connected to intrinsic features or to extrinsic measures?

Both in literature and within the PR&PP WG the issue of characterising Safeguardability as an intrinsic or an extrinsic characteristic of a nuclear energy system emerged. In the Proliferation Resistance domain these two concepts (intrinsic and extrinsic) were defined in [15]. Intrinsic proliferation resistance features are *those features that result from the technical design of nuclear energy systems, including those that facilitate the implementation of extrinsic measures*[4], and extrinsic proliferation resistance measures are *those measures that result from States' decisions and undertakings related to nuclear energy systems*[5] .

When reasoning on the Safeguardability concept, one possibility might be to define it as the collection of those features resulting by the technical design of the system, that facilitate the implementation of extrinsic proliferation resistance measures. This definition would classify Safeguardability as an intrinsic proliferation resistance feature.

Although the above point of view might well be adopted, it is worthwhile to take a broader and holistic point of view, and (following [19]) begin to acknowledge that any technical (hard) system is embedded (integrated) in a human, social (soft) system in which it is operated and with which it interacts in numerous and often complex ways. Indeed, failures leading to catastrophic accidents often occur due to failures or misunderstandings or understatements of these interactions between hard and soft systems. In our case, the nuclear energy system is deployed and operated by an operator, and

[2] [3], p.66.
[3] For a more detailed description of the holon concept in technical areas, see [19].
[4] [15], p.1.
[5] [15], p.2.

subject to international non proliferation agreements leading to nuclear safeguards verification activities. Once this scenario is accepted, Safeguardability could be defined as a property emerging from the interaction of a nuclear energy system (hard) with the activities connected with the non proliferation legal framework in force (soft). In particular, Safeguardability could be seen as an index of the *potential quality* of this interaction.

### 5.3. Safeguardability and Proliferation Resistance: a tight and complex relationship

The relationship between the Safeguardability concept and Proliferation Resistance is certainly a tight one, but its characterisation strongly depends on how Safeguardability and Proliferation Resistance are defined.

If the definition of Proliferation Resistance given in [15] is accepted, and the first possibility of defining Safeguardability given in 4.1 is assumed, then Safeguardability would be seen as a sub set of the PR intrinsic features.

If we adopt an holistic point of view, and accept that hard systems are integrated in soft systems with which they interact, Proliferation Resistance and Safeguardability could be seen as indexes of the potential quality of different interactions of the hard system with the soft system: Proliferation Resistance could be seen as measure of the potential quality of the interaction of a nuclear energy system with activities connected with a proliferation effort, and Safeguardability, as stated in the previous paragraph, could be seen as a measure of the potential quality of the interaction of the nuclear energy system with the activities connected with the non proliferation legal framework in force.

It is worth noticing that some of the attributes identified for analysing Safeguardability are relevant also for analysing proliferation resistance, but their contribution might be a positive one in one case and a negative one in the other. For example, limited accessibility to nuclear material due to the radiological hazards connected to the radiation field is negative for Safeguardability (inspectors' activities are negatively affected), but very positive for proliferation resistance (a good radiological barrier increases the technical difficulty associated to a diversion scenario). This aspect puts in evidence that designing a nuclear energy system excelling in ensuring non proliferation is a challenging task, where trade-offs on a number of important aspects will have to be achieved, and optimisation of these trade-offs will not be always straightforward.

## 6. Conclusions

Due to various reasons, there is a renewed interest in nuclear energy as an important player in the near to mid future. This led to various international efforts aimed at shaping and designing future nuclear power plants and their related fuel cycle. Among other goals, future nuclear energy systems will have to be proliferation resistant and will have to operate under an international nuclear safeguards regime.

Nuclear Energy Systems designers repeatedly asked the GIF PR&PP WG to have guidelines to ensure that their teams take the issue of Safeguardability into account at early design stages, and the group is developing this concept for answering to this need. As a first step of the work, list of relevant attributes has been developed and published in the latest revisions of the GIF PR&PP Evaluation Methodology Report.

Work on the subject is still ongoing and currently an external validation process and a further characterisation of the identified attributes is being performed at JRC, taking advantage of JRC experts in the relevant domains. The proposed way forward for this activity has been presented in section 4.

While continuing to contribute to the main stream of the PR&PP WG activities, JRC is exploring the possibility to couple the experience on the concepts of Proliferation Resistance and Safeguardability gained in the GIF PR&PP frame with the internal know-how on Safeguards and Non-proliferation to try develop a holistic approach for analysing the evidence that a nuclear energy system design will be highly safeguardable. This activity is being carried out adopting a holistic systems thinking approach developed at the University of Bristol (UK).

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# GIF Evaluation Methodology for Proliferation Resistance and Physical Protection of Generation IV Nuclear Energy Systems: Overview and Perspectives

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***Abstract:***

*The Generation IV International Forum (GIF) was initiated in 2000 and formally chartered in mid 2001. It was set-up as an international collective representing the governments of ten Countries (Argentina, Brazil, Canada, France, Japan, South Korea, South Africa, Switzerland, UK and the USA) strongly involved in the deployment and development of nuclear technology for energy production. The European Atomic Energy Community (EURATOM), represented by the European Commission, signed the GIF agreement on July 30, 2003.*

*The Technology Goals for Generation IV nuclear energy systems, developed during the Roadmap project [1], highlight Proliferation Resistance and Physical Protection (PR&PP) as one of the four goal areas for these technologies, along with Sustainability, Safety & Reliability, and Economics. On the basis of these four goal areas an evaluation methodology was developed which contributed to identify the six nuclear energy systems options currently under consideration by GIF.*

*The Generation IV Roadmap recommended the development of a comprehensive evaluation methodology to assess PR&PP of Generation IV nuclear energy systems. Accordingly the PR&PP Working Group was formed and tasked by the GIF in December 2002 to develop an improved evaluation methodology on the basis of the Roadmap's recommendation. The expert group includes members of the GIF and representatives from the IAEA.*

*The methodology is organised as a progressive approach applying alternative methods at different levels of thoroughness as more design information becomes available and research improves the depth of technical knowledge. To date, the overall framework of the methodology is considered rather accepted and stable; the methodology was advanced with a development case study and has been tested though a demonstration case study.*

*This paper provides an updated overview of the methodology approach developed by the PR&PP Working Group. The paper also highlights some of the achievements and the lessons learned during the demonstration case study in which different techniques have been applied for the implementation of the PR&PP evaluation approach. Finally the paper presents some of the future directions for the activity of the group.*

**Keywords:** Proliferation Resistance, Physical Protection Robustness, Evaluation/Assessment.

## 1. Introduction

The Technology Goals for Generation IV nuclear energy systems (NESs) highlight Proliferation Resistance and Physical Protection (PR&PP) as one of the four goal areas along with Sustainability, Safety and Reliability, and Economics [1]:

Generation IV nuclear energy systems will increase the assurance that they are a very unattractive and the least desirable route for diversion or theft of weapons-usable materials, and provide increased physical protection against acts of terrorism.

Proliferation resistance is that characteristic of an NES that impedes the diversion or undeclared production of nuclear material or misuse of technology by the Host State seeking to acquire nuclear weapons or other nuclear explosive devices.

Physical protection (robustness) is that characteristic of an NES that impedes the theft of materials suitable for nuclear explosives or radiation dispersal devices (RDDs) and the sabotage of facilities and transportation by sub-national entities and other non-Host State adversaries.

The Generation IV Roadmap recommended the development of an evaluation methodology to assess NESs with respect to PR&PP. Accordingly, the Generation IV International Forum formed an Expert Group in December 2002 to develop a methodology. The development of methodology has been documented in a number of reports [2-7].

This paper provides an updated overview of the methodology approach developed by the PR&PP Expert Group [8, 9]. The paper also highlights some of the achievements and the lessons learned during the demonstration case study in which different techniques have been applied for the implementation of the PR&PP evaluation approach [10]. Finally the paper presents some of the future directions for the activity of the group.

## 2. Overview of PR&PP Methodology

In this section a brief overview of the PR&PP methodology is given on the basis of the executive summary of the Revision 5 methodology report [8]: figure 1 illustrates the PR&PP methodological approach at its most basic. For a given system, analysts define a set of challenges, analyze system response to these challenges, and assess outcomes. The challenges to the NES are the threats posed by potential proliferant States and by sub-national adversaries. The technical and institutional characteristics of the Generation IV systems are used to evaluate the response of the system and determine its resistance to proliferation threats and robustness against sabotage and terrorism threats. The outcomes of the system response are expressed in terms of PR&PP measures and assessed.

The evaluation methodology assumes that an NES has been at least conceptualized or designed, including both the intrinsic and extrinsic protective features of the system. Intrinsic features include the physical and engineering aspects of the system; extrinsic features include institutional aspects such as safeguards and external barriers. A major thrust of the PR&PP evaluation is to elucidate the interactions between the intrinsic and the extrinsic features, study their interplay, and then guide the path toward an optimized design that identifies and minimizes vulnerabilities.

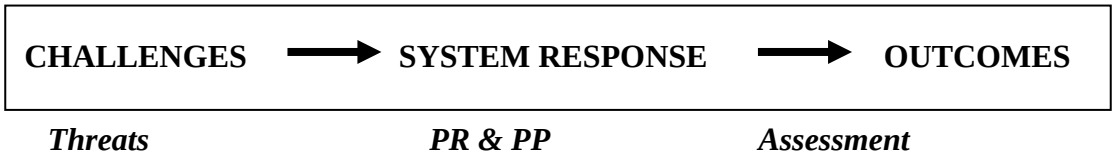


Figure 1: Basic Framework for the PR&PP Evaluation Methodology.

The structure for the PR&PP evaluation can be applied to the entire fuel cycle or to portions of an NES. The methodology is organized as a progressive approach to allow evaluations to become more detailed and more representative as system design progresses. PR&PP evaluations should be performed at the earliest stages of design when flow diagrams are first developed in order to systematically integrate proliferation resistance and physical protection robustness into the designs of Generation IV NESs along with the other high-level technology goals of sustainability, safety and reliability, and economics. This approach provides early, useful feedback to designers, program policy makers, and external stakeholders from basic process selection (e.g., recycling process and type of fuel), to detailed layout of equipment and structures, to facility demonstration testing. Figure 2 provides an expanded outline of the methodological approach. The first step is threat definition. For both PR

and PP, the threat definition describes the challenges that the system may face and includes characteristics of both the actor and the actor's strategy. For PR, the actor is the Host State for the NES, and the threat definition includes both the proliferation objectives and the capabilities and strategy of the Host State. For PP threats, the actor is a sub-national group or other non-Host State adversary. The PP actors' characteristics are defined by their objective, which may be either theft or sabotage, and their capabilities and strategies.

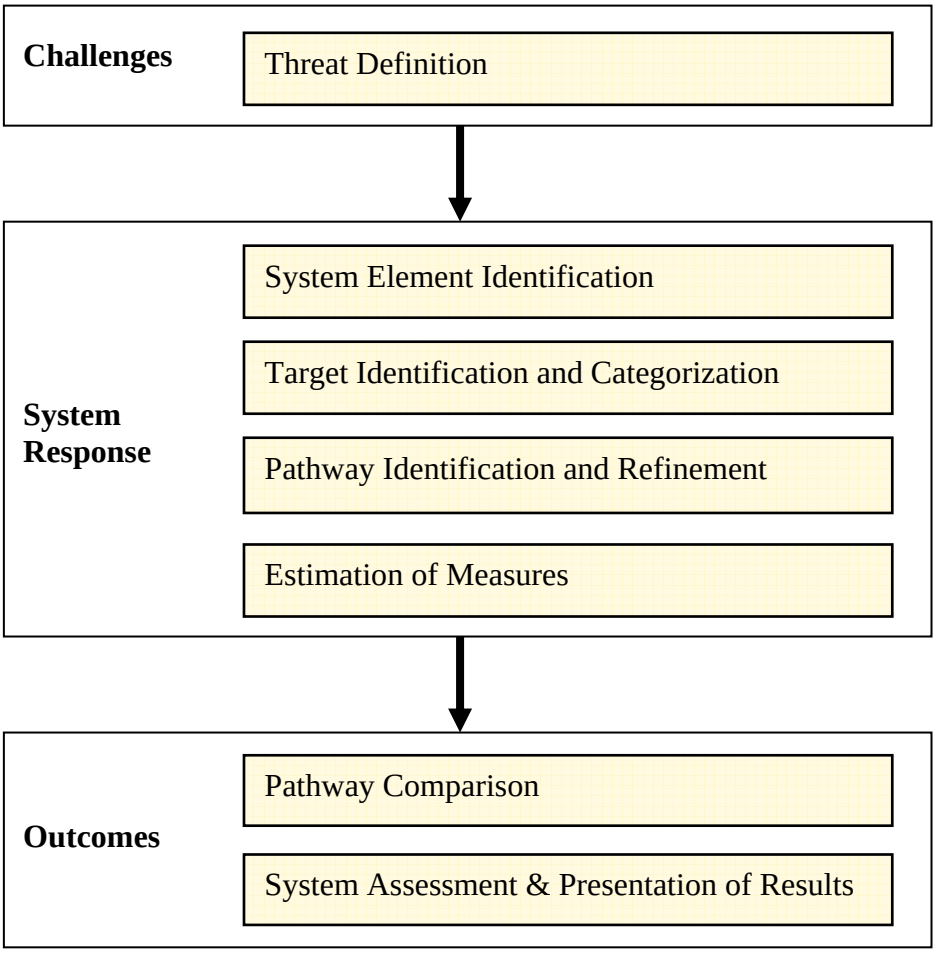


Figure 2: Detailed Framework for the PR&PP Evaluation Methodology.

To facilitate the comparison of different evaluations, a standard Reference Threat Set (RTS) can be defined, covering the anticipated range of actors, capabilities, and strategies for the time period being considered. Reference Threat Sets should evolve through the design and development process of nuclear fuel cycle facilities, and for physical protection ultimately becoming Design Basis Threats (DBTs) upon which regulatory action is based.

For PR, the threats include:

- Concealed diversion of declared materials;
- Concealed misuse of declared facilities;
- Overt misuse of facilities or diversion of declared materials;
- Clandestine dedicated facilities.

For PP the threats include:

- Radiological sabotage;
- Material theft;

- Information theft.

The PR&PP methodology does not determine the probability that a given threat might or might not occur. Therefore, the selection of what potential threats to include is performed at the beginning of a PR&PP evaluation, preferably with input from a peer review group organized in coordination with the evaluation sponsors. The uncertainty in the system response to a given threat is then evaluated independently of the probability that the system would ever actually be challenged by the threat. In other words, PR&PP evaluations are contingent on the challenge occurring.

The detail with which threats can and should be defined depends on the level of detail of information available about the NES design. In the earliest stages of conceptual design, where detailed information is likely limited, relatively stylized but reasonable threats must be selected. Conversely, when design has progressed to the point of actual construction, detailed and specific characterization of potential threats becomes possible.

When threats have been sufficiently detailed for the particular evaluation, analysts assess system response, which has four components:

1. System Element Identification. The NES is decomposed into smaller elements or subsystems at a level amenable to further analysis. The elements can comprise a facility (in the systems engineering sense), part of a facility, a collection of facilities, or a transportation system within the identified NES where acquisition (diversion) or processing (PR) or theft/sabotage (PP) could take place.
2. Target Identification and Categorization. Target identification is conducted by systematically examining the NES for the role that materials, equipment, and processes in each element could play in each of the strategies identified in the threat definition. PR targets are nuclear material, equipment, and processes to be protected from threats of diversion and misuse. PP targets are nuclear material, equipment, or information to be protected from threats of theft and sabotage. Targets are categorized to create representative or bounding sets for further analysis.
3. Pathway Identification and Refinement. Pathways are potential sequences of events and actions followed by the actor to achieve objectives. For each target, individual pathways are divided into segments through a systematic process, and analyzed at a high level. Segments are then connected into full pathways and analyzed in detail. Selection of appropriate pathways will depend on the scenarios themselves, the state of design information, the quality and applicability of available information, and the analyst's preferences.
4. Estimation of Measures. The results of the system response are expressed in terms of PR&PP measures. Measures are the high-level characteristics of a pathway that affect the likely decisions and actions of an actor and therefore are used to evaluate the actor’s likely behavior and the outcomes. For each measure, the results for each pathway segment are aggregated as appropriate to compare pathways and assess the system so that significant pathways can be identified and highlighted for further assessment and decision making.

For PR, the measures are:

- Proliferation Technical Difficulty (TD) – The inherent difficulty, arising from the need for technical sophistication and materials handling capabilities, required to overcome the multiple barriers to proliferation.
- Proliferation Cost (PC)– The economic and staffing investment required to overcome the multiple technical barriers to proliferation including the use of existing or new facilities.
- Proliferation Time (PT) – The minimum time required to overcome the multiple barriers to proliferation (i.e., the total time planned by the Host State for the project).
- Fissile Material Type (MT) – A categorization of material based on the degree to which its characteristics affect its utility for use in nuclear explosives.
- Detection Probability (DP) – The cumulative probability of detecting a proliferation segment or pathway.
- Detection Resource Efficiency (DE) – The efficiency in the use of staffing, equipment, and funding to apply international safeguards to the NES.

For PP, the measures are:

- Probability of Adversary Success (PS) – The probability that an adversary will successfully complete the actions described by a pathway and generate a consequence.
- Consequences (C) – The effects resulting from the successful completion of the adversary's action described by a pathway.
- Physical Protection Resources (PPR) – the staffing, capabilities, and costs required to provide PP, such as background screening, detection, interruption, and neutralization, and the sensitivity of these resources to changes in the threat sophistication and capability.

By considering these measures, system designers can identify design options that will improve system PR&PP performance. For example, designers can reduce or eliminate active safety equipment that requires frequent operator intervention.

The final steps in PR&PP evaluations are to integrate the findings of the analysis and to interpret the results. Evaluation results should include best estimates for numerical and linguistic descriptors that characterize the results, distributions reflecting the uncertainty associated with those estimates, and appropriate displays to communicate uncertainties.

The information is intended for three types of users: system designers, program policy makers, and external stakeholders. Thus, the analysis of the system response must furnish results easily displayed with different levels of detail. Program policy makers and external stakeholders are more likely to be interested in the high-level measures, while system designers will be interested in measures and metrics that more directly relate to the optimization of the system design.

## 3. The Demo Case Study: Achievements and Lessons Learned

This section summarizes some of the results achieved during the Demonstration Case study which was carried out by the group in the period 2005-2006. The text is mainly based on the Executive Summary of the Demonstration Study Interim Report [10].

### 3.1 Introduction

The Demonstration Study aimed at demonstrating the application of the framework for proliferation resistance (PR) evaluation, as being developed by the PR&PP Working Group, to elements of a nuclear energy system. Three evaluation approaches, the qualitative evaluation approach, the event tree/fault tree approach, and the Markov approach, are each applied by a different task group of the PR&PP Working Group to evaluate the PR measures. Selected for the demonstration study is a portion, or "slice," of the fuel cycle facility (FCF) for the Example Sodium Fast Reactor (ESFR). The FCF being modeled is a pyro-chemical reprocessing (pyroprocessing) facility designed to accept the spent sodium-bonded, metallic fuel from four advanced fast reactors and to convert it into three output streams (new fuel assemblies, metal waste ingots, and ceramic waste forms). Pyroprocessing is a process that separates uranium, transuranics, and fission products using electrochemically driven transport between molten salt and metal phases. Under normal operation, the processes do not separate plutonium from the minor actinides, and therefore, all material handling occurs remotely in hot cells, where personnel access does not occur except under highly special circumstances.

### 3.2 The Pyproprocessing Fuel Cycle Facility

The pyroprocessing technology as applied in the facility for the demonstration study has five main process steps.

1. Spent fuel assemblies are disassembled and the resulting fuel elements are mechanically chopped.
2. Chopped elements are electro-refined to partially separate the uranium from fission products and actinide elements. This step generates a uranium material, which is further processed to remove adhered salt and produce the uranium (U) product. This second step also generates metal waste resulting from undissolved cladding hull pieces.

3. This step consists of recovering the transuranic (TRU) material that is present in the salt used for uranium electro-refining. Similar to the uranium material, TRU/U material recovered is further processed to remove adhered salt and produce the TRU/U product.
4. The U product, TRU/U product, and fissile makeup material are melted together to produce fuel slugs. Fuel elements are then fabricated from these slugs and assembled into fuel assemblies to be returned to the co-located reactors. In this step, external material from Light Water Reactors (LWR) (Uranium and External TRU metal) is added to the process.
5. The final step consists of conditioning the metal and salt wastes generated by the second and third steps, respectively, and producing ceramic/metal waste forms for disposal.

Figure 3 shows the process steps, the material flows and includes elements of the safeguards system.

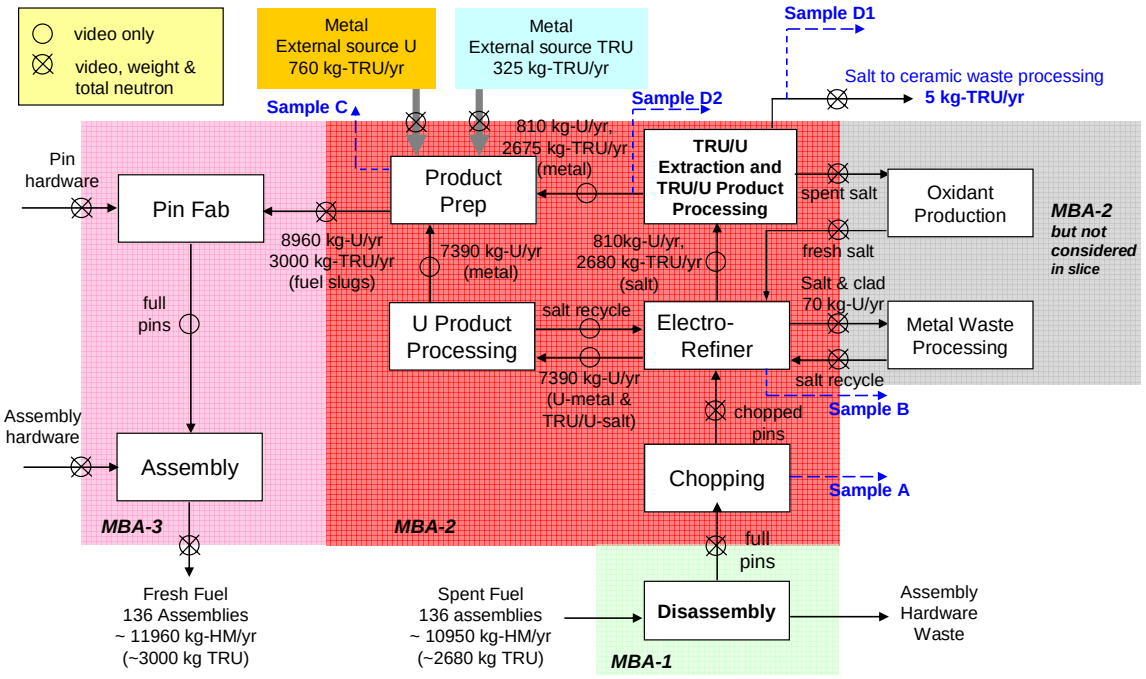


Figure 3. Process steps, material flows and elements of the safeguard system.

### 3.3 The Safeguards Approach

Three Material Balance Areas[1] (MBAs) are defined for the demonstration slice.

- MBA-1: The spent fuel element disassembly process occurs in the Receiving/Shipping Cell. In this MBA, spent fuel assembly items are disassembled into smaller spent fuel element items.
- MBA-2: Electrochemical processing occurs in the Process Cell.
- MBA-3: New fuel manufacture occurs in the Fresh Fuel Hall and the Assembly Fabrication that occurs in the Receiving/Shipping Cell.

It is assumed that the safeguards controls to be installed around the MBAs will use neutron counters, cameras, seals, and a material accountability system based on an initial evaluation of the Cm/Pu ratio performed on the spent fuel as it enters the facility.

[1] According to IAEA Safeguards Glossary, 2001 Edition, an MBA is defined, according to IAEA INFCIRC-153, as "*an area in or outside of a facility such that: (a) The quantity of nuclear material in each transfer into or out of each 'material balance area' can be determined; and (b) The physical inventory of nuclear material in each 'material balance area' can be determined when necessary, in accordance with specified procedures, in order that the material balance for Agency safeguards purposes can be established*".

### 3.4. The Threat and the Evaluations

The assumed proliferation scenario is that of a host-state diversion. For this study, the threat definition corresponds to the characteristics generally considered for a reactor state with technical know-how and industrial infrastructure. The host state is assumed to be a non-nuclear weapons state (NNWS), a signatory to the NTP and have an Additional Protocol (AP) in force. The objective of the proliferators is to divert covertly 1 significant quantity (SQ) equivalent of nuclear material from the FCF within one year without detection by safeguards and process the diverted nuclear material in clandestine facilities.

The paradigm for the proliferation resistance methodology, being developed by the PR&PP Expert Group, is composed of three elements (see figure1). In this paradigm, for a given system a set of challenges is identified; the system response to these challenges is analyzed, and outcomes are determined.

The system response involves:

- Subdividing the nuclear system into 'elements' (a facility, part of a facility, a collection of facilities, or a transportation system);
- Identifying potential targets within each element (targets are the nuclear material and processes to be protected from PR threats ), and
- Identifying and evaluating all potential sequences of events (Pathway Analysis) that could result in the proliferant state succeeding in its objective of concealed diversion of 1 SQ.

The outcomes of the pathways are expressed in terms of PR measures:

- Proliferation Technical Difficulty;
- Proliferation Cost;
- Proliferation Time;
- Fissile Material Type;
- Detection Probability;
- Detection Resources Efficiency.

The major differences among the three evaluation approaches used in this demonstration study are in the implementation of the evaluation steps (identify system elements, identify targets, and identify and evaluate sequences of events). As a result the quantification of the PR measures undergoes a different process for the three evaluation approaches. The qualitative approach relies on expert judgment to assign values and uncertainty ranges to PR measures, using a set metric scales for the PR measures[2]. Not all measures are amendable to the event tree/fault tree approach (e.g. proliferation time). In that case auxiliary methods are needed to quantify the remaining measures. The Markov approach also cannot quantify all PR measures directly without resort to auxiliary calculations; one example is detection resources efficiency.

#### 3.4.1 The Qualitative Evaluation

All quantitative PR&PP evaluations necessarily start with a qualitative one. Qualitative evaluation provides a necessary tool to structure the analysis problem and prioritize areas for detailed study using quantitative methods. In applying the qualitative evaluation approach to the demonstration slice, the emphasis is on the importance of using a structured approach in performing qualitative evaluation; the study also presents guidelines and examples for this process.

Steps involved in a qualitative evaluation include:

1. Select system elements and the threat for study
    - Gather design information, may range from conceptual to detailed
    - Specify objectives, capabilities, and strategies (can be stylized descriptions) for the selected threat

[2] Example scales for the estimates of each of the PR measures are reported in [8].

2. Use expert judgment to survey system elements and threat strategies to identify a small number of “representative” pathways for analysis
3. Perform qualitative analysis using expert judgment to estimate measure values for selected pathways.
    - Consider acquisition and processing separately, then aggregate measures
    - Use check lists, as available, to assure that important system attributes have been considered
    - Display results in tabular form, showing uncertainty intervals as ranges for the metrics for each measure
4. Use insights from (3) to confirm initial selection in (2).
5. Discuss insights and conclusions from analysis.

PR measures are evaluated by expert judgment. The scales for the PR measures range from very high (VH) PR, making the pathway less attractive to a proliferant state, to very low (VL) PR, making the pathway more attractive to the state. Uncertainty bands are estimated for each measure. A qualitative uncertainty band is used to reflect the state of early phase of design and analysis when incomplete information is available. A narrower uncertainty band, the residual uncertainty band, is used to represent potential uncertainty range after detailed design and analysis are completed.

A total of four diversion scenarios are considered in the qualitative evaluation. The first three diversion scenarios assumed protracted diversion by the host state from different unit operations within the MBA-2. The diversions are from:

- The spent fuel chopping operation;
- The TRU extraction operation;
- The product preparation operation.

The fourth diversion scenario is for a distributed diversion strategy, i.e., removing material clandestinely from many parts of the pyroprocessing facility. This pathway involves protracted, concealed diversion of material from the facility, aimed at acquiring TRU without detection by safeguards. This TRU is subsequently processed in a separate, concealed facility to produce plutonium metal for fabrication into nuclear explosives.

The PR measures for the four diversion scenarios have been evaluated by two experts using the qualitative approach. The first three scenarios of concentrated diversion were evaluated by one expert (LLNL) and the distributed diversion scenario is evaluated independently by the other expert (UCB). In evaluating Technical Difficulty (TD), the experts considered both intrinsic and extrinsic barriers, worker skill, and industrial capability of the host state. The TD measure ranged from low to medium-high for the four diversion scenarios. Proliferation time (PT) depends on the diversion rate and the need to have access to clandestine processing facilities. PT was judged to hover around medium, roughly 1 to 5 years. Proliferation cost (PC) can vary from low to high depending on the type and rate required to process the diverted material in clandestine facilities. Fissile Material Type (MT) is a measure to be estimated on the material available at the end of the processing stage and, therefore, is estimated once for the overall pathway. A qualitative measure for MT is the attractiveness or usability of the material for weapons. For different blends of TRU coming out of the pyroprocessing facility MT was judged to range from medium to medium-high. Considering the various safeguards approaches available to detect diversion and the type of operation associated with each diversion point, Detection Probability (DP) for the four scenarios was ranged from low to high. Detection Resources Efficiency (DE) is evaluated against IAEA inspector efforts for similar facilities. Based on estimated effort for the pyroprocessing facility (scaled from effort for PUREX plants) DE was judged to be low to very low.

### 3.4.2 The Logic Trees Based Evaluations

Two separate but complementary tree-based evaluations were taken by teams from PNNL and MIT in this methodology demonstration. The first (by PNNL) used fault trees to model potential failure of an attempt at diversion, the second (by MIT) used success trees to model the likelihood of success of such an attempt. The two teams worked closely together to ensure that the basis for the two separate analyses was comparable. Either approach can be used; the choice depends on the problem being studied and the preference of the analyst. The fault/success tree study only evaluates the material

acquisition phase from the facility slice. Due to funding and time constraints, the study was not completed, but sufficient progress was made to demonstrate the utility of the methodology.

The application of the event tree/fault tree approach to the evaluation of proliferation resistance is a three step process. The first step is to do a threat analysis and identify potential diversion points or pathways by way of an event tree analysis. Event trees are inductive logic models used to identify sequences of events that lead to particular outcomes, both desirable and undesirable. The second step is to define a diversion strategy and identify corresponding safeguards detection methods to be overcome by the proliferators. A fault tree structure is constructed to model the failure of each safeguards detection methods to detect potential diversion attempts within a specified time. Fault trees are deductive logic models constructed to define all possible failure combinations which lead to a particular event, for instance the failure of a specific system to function as required to perform a vital mission. The third and final step is to evaluate the likelihood of detection of the proliferation attempt and to calculate the six proliferation resistance measures. The solution of the event tree/fault tree models is a collection of what are called minimal cutsets. Minimal cutsets are combinations of occurrences (basic events) along a pathway that allow the pathway consequence to occur. Each minimal cutset has a probability based on the concatenation of the likelihood of each base event or occurrence in the minimal cutset.

Based on analyzing the operation of the pyroprocessing facility and the daily material flow through the demonstration slice it was decided to evaluate the likelihood of proliferation success for a protracted strategy using the External Uranium Container in the Product Preparation Station in MBA-2. Since the container has been described to have a capacity to carry 3.17 kg of uranium it is assumed in the pilot study that the diversion strategy will be to perform three diversions in one year. Each diversion attempt will consist of 3.17 kg of TRU metal from the Product Preparation station. A fault tree is prepared for each diversion attempt. For this pilot study screening values were used to determine the likelihood of each basic event (occurrences along a pathway). The enablers (actions taken by the facility owner to defeat safeguards) were given a probability of 1.0; the failures of safeguards personnel were estimated based on human failure probabilities and assigned a screening value of either 2.5E-1 or 5.0E-1, and the failures of the instruments to detect the diversion were also set at screening values of either 2.5E-1 or 5.0E-1. If the analysis had proceeded further, more detailed values would have been developed.

It should be noted that minimal cutsets are considered “raw data” by analysts, and they must be reviewed for validity and plausibility. If this study had been completed, the minimal cutsets would have been grouped to represent specific diversion scenarios for each pathway, and these scenarios would have been developed to provide:

- Proliferation Detection Probability;
- Proliferation Pathway Technical Difficulty based on minimal cutset evaluation;
- Proliferation Pathway Resources Efficiency would have been based on minimal cutset evaluation.

Material Type would have been developed based on the condition of the material being diverted. In the case of the pathway being analyzed, since the diverted material was TRU metal that had gone through the electrorefining process, the material type would have had value between reactor grade plutonium and deep burn grade plutonium. Proliferation Time in this model was assumed to be one year; the Detection Resources Efficiency would have been determined by an evaluation of the cost of the safeguards involved and the personnel costs required to support them.

A complementary tree-based method, the success tree approach is also used to implement the pathway analysis of the PR evaluation methodology in this study. The diversion scenario of weapons material (Pu is assumed in this study) is modeled by discrete steps at MBA-2 of the FCF. The diversion scenario is divided into four steps: (1) lying about the amount of the input material, (2) moving Pu to holdup inside the MBA-2, (3) moving Pu from holdup to baskets going out of the ‘Product Prep (PP)’ stage, and finally (4) diverting Pu via the PP basket.

A multi-step diversion is assumed in the success tree analysis to get 1 SQ of Pu, because 1 SQ is too much to divert safely in a single attempt, and without being detected. 10 diversion attempts to obtain 1 SQ and 3 attempts to obtain 1 SQ are modeled respectively as examples. Dependencies between

each step diversion are assumed. For example, the success of the nth diversion step depends on the success of the (n-1)st diversion step. This treatment is based upon the inference that manipulations of surveillance equipment such as surveillance cameras are needed in order to fool the cameras, for example, in order to move Pu inside MBA-2 without being detected. If this manipulating or fooling of the cameras is repeated in order to get 1 SQ of Pu, then the possibility of this manipulation being detected should be increased remarkably because of multiple occurrences. On the other hand, safeguard inspectors might become complacent after seeing repeated occurrences and the possibility of being detected could actually go down for multi-step diversion.

Dependencies are assumed for the activities in this success tree model as follows:

1. Lie -> No dependency between the diversion steps is assumed;
2. Move Pu to holdup -> Fooling cameras inside MBA-2 is needed. -> Dependency exists;
3. Move Pu from holdup -> Fooling cameras inside MBA-2 is needed. -> Dependency exists; and
4. Divert Pu via PP outgoing baskets. -> Fooling cameras monitoring the baskets is needed. -> Dependency exists.

Sensitivity analysis done on the dependency of failure probability on previous diversion attempts found that the dependency has negligible effect on the final result, the proliferation success probability.

### 3.4.3 The Markov Evaluation

The Markov model approach is an implementation of the pathway analysis, a key element of the PR&PP evaluation methodology being developed by the PR&PP Expert Group. The Markov chain method has the capability to account for some of the dynamic features of proliferation, namely the large number of uncertainties, the unpredictability of human performance, and the effect of changing conditions with time. In the Markov model approach the normal flow of nuclear material in the fuel cycle (front and back ends) are accounted for and the abnormal flow due to proliferation activities are modeled as a time dependent random process. Major activity modules in the fuel cycle (e.g., a physical process in a recycle facility) and the proliferation pathway (e.g. the act of diversion from a declared facility) are represented by a number of discrete stages in the Markov chain. In addition, absorbing states (terminal stages) are used to represent the effective termination of the proliferation activity due to intrinsic (e.g., radiation) or extrinsic (e.g., international safeguards) barriers. The transition between stages is treated as a random process with a given probability distribution. The transition rate is characterized by time parameters that are based on physical processes. For example the transition time from one process to the next in the fuel cycle facility is derived from the rate of material flow in the actual recycling process. In modeling safeguards the rate of detecting an anomaly is derived from the frequency of executing safeguards approaches. The realization of the random process at each stage is then a random variable and the expected values of these random variables constitute the state (solution) space. Thus by mapping the stages of a proliferation scenario into a Markov chain model the likelihood of all possible outcomes can be determined systematically.

The Markov model approach is highly adaptable and scaleable. It has been applied previously to evaluate the PR of an advanced light water reactor in a misuse scenario and in scenarios that involve diversion from the front and back ends of a once through fuel cycle. In the demo study the Markov approach was applied to evaluate the PR measures for a portion of the fuel cycle facility of the ESFR system. As a result of an assessment of the system response to the threats, probabilistic PR measures such as detection probability and failure probability (due to intrinsic barriers and technical difficulties) are calculated directly by the Markov model. Since the model also represents the pathways for which the time and cost parameters can be defined, the path with least cost or the least time can also be calculated using the same model. In addition, the type of diverted material (pathway dependent), proliferation cost, and detection resources can all be estimated based on the pathway analysis.

The demonstration study considers the case of a host-state diversion and the aspiration is to obtain 1 SQ equivalent of TRU in one year. The Markov model is applied to evaluate two diversion strategies and two tactics. The strategies are concentrated (from one place) and distributed diversion, and the tactics are abrupt and protracted diversion. Safeguards approaches considered in the Markov model include audit of nuclear material accounting reports and records, material verification, surveillance and monitoring, and containment. The level of complexity of the Markov model for PR evaluation has

increased over the past year. New features have been added by introducing new parameters that influence the values of the transition times. In particular, the following new features are introduced in the demonstration slice study:

1. An effective detection rate has been introduced to account for the implementation of multiple safeguards approaches at a given strategic point. Uncertainties related to the accuracy/sensitivity of measurement methods are considered in the model. The potential for false alarm due to over-sensitivity of safeguards equipment is accounted for by a new parameter, the confidence level of diversion confirmation.
2. A new state called “diversion failure” is introduced to reflect the inability of the proliferators to overcome the intrinsic barriers originated from either the design of the facility or the properties of the material in the facility.
3. Concealment to defeat or degrade the performance of safeguards is recognized in the Markov model. It is considered as a tactic of the proliferators and is assumed to prompt more immediate and concerted responses from the safeguards inspectors.
4. Human performance in the safeguards area is incorporated in the Markov model by modifying the time parameter of a human action (e.g. the transition time associated with an inspection) with a success factor that takes into consideration the probability of human errors.

Example cases have been done to demonstrate the effects of modeling features on the PR measures for the fuel facility of the ESFR system. One of the safeguards approaches observed to have a positive impact on successful detection of diversion activities is the proper employment of surveillance cameras because they are able to detect an anomaly quickly. The effects of false alarms on detection probability, failure probability and success probability have been studied. With the presence of intrinsic barriers, a new absorbing state is introduced, diversion failure due to intrinsic barriers. This new absorbing state has several effects on the outcome of the proliferation activity. It reduces the probability of detection because now there are two failure terminal states, being detected and diversion failure due to intrinsic barriers. The presence of intrinsic barriers is demonstrated to have the effects of decreasing the transition rate from declared facilities to clandestine facilities and also prolonging the proliferation time. Though the detection probability is lower with intrinsic barriers, the overall success probability for the proliferator is lower due to significant increase in diversion failure from intrinsic barriers. Concealment enables the proliferator more chance to divert material and also fail in the attempt due to intrinsic barriers. Concealment reduces the detection probability and increases the diversion failure probability and the success probability for the proliferator. The impact of human errors is similar to that of concealment. Results of the analysis indicate that human errors have the effects of lowering the detection probability slightly while enabling the proliferators to have more chance to divert material and fail for the same intrinsic barriers. The case of distributed diversion is compared with concentrated diversion for the demonstration slice. The results indicate a lower probability of detection and shorter proliferation time for the distributed case.

Three of the six PR measures are calculated directly by the Markov model and they are the detection probability (DP), proliferation technical difficulty (TD) and proliferation time (PT). Technical difficulty can occur in overcoming intrinsic barriers or in processing the diverted material. A metric PF, probability of technical failure, is used as quantitative realization of the measure for proliferation technical difficulty. The other three measures are derived based on the material type. Material type (MT) is indicated by an index that is based on the type of material at the acquisition stage. By assessing the physical, chemical, and isotopic properties of the diverted material the proliferation cost (PC) and resources required to detect the proliferation can then be evaluated. Inn the Demo study, PC is evaluated according to the easiness of converting the diverted material to Pu, and so, it is dependent on MT (both isotopic composition and quantity). Detection resources efficiency (DE) is also tied to MT because more resources will be allocated to protect materials that are of interest and use to the proliferators. The differences in detection probabilities are shown not to be very large among the facilities in the demonstration slice. Results from the Markov model suggest that among the 8 facilities, considered in the study, U-product processing and electro-refiner are the most proliferation resistant, while TRU extraction is the least proliferation resistant. The main reason for being the most or least proliferation resistant is in the attractiveness of the material (TRU salt versus TRU metal). The presence of significant intrinsic barriers in the material balance area (e.g. operation in hot cell) also has some bearing on the relative resistance to proliferation when comparing different facilities.

Sensitivity analysis for the time parameters for the Markov model has been performed. It is noted that variation in one parameter affects several measures. This behavior is consistent with the fact that the PR measures are not independent of each other. Results of the sensitivity analysis show that the overall impact of intrinsic barriers is significant for PT and PF and minor for the detection probability. For the designer of a facility, once the recycling process is determined there is little that can be done to alter the MT of the material in the facility. At the assumed baseline level of detection capability, the overall benefit of increasing safeguards is not obvious from the sensitivity analysis. There seems to be an indication of diminishing return in the cost related to increasing safeguards (increased capability and frequency). Evaluating the impact of diversion rate on proliferation resistance is less conclusive because of uncertainties and assumptions in the safeguards approaches. While decreasing the diversion rate will exploit the uncertainties of the safeguards such that the chances of being detected will decrease, it will also prolong the diversion process and increases the detection probabilities and proliferation time. Sensitivity analyses have been performed to evaluate the impact of increased and decreased diversion rates on the PR measures.

## 4 Accomplishments and Future directions

The demonstration study showed that the methodology developed by the GIF PR&PP Working Group provides a structured framework for comprehensive evaluation of the PR for a nuclear system. The PR measures provide a means for comparison of key characteristics of each pathway. Results from the three different forms of evaluation (qualitative, event tree, Markov model) were consistent, although the level of detail and focus of each analysis differed. These differences were due in part to the focus of the analysts and in part to the choice of analytic methods.

In summary, qualitative evaluation is well suited to coarse-level evaluation of a nuclear system where detailed information about the system is not available and results are required quickly. More detailed analysis using methods such as logic tree or Markov modeling are appropriate where more information about the nuclear system is available and where more accurate results (lower uncertainty) are required. Further analyses on diverse nuclear systems with diverse analytic objectives (e.g. quick study for decision maker; detailed study for designer) should be conducted to gain further insight into appropriate application of analytic methods to PR&PP analysis and to establish a baseline standard/norm for such analysis.

In the year 2007-08, the PR&PP Working group plans to carry out a broader fuel cycle study, involving different threats. Strategies to be considered will be: Diversion, Misuse and Abrogation for PR and theft for PP.

Among the methodological aspects to be further investigated, there will be the need to develop systematic approaches for expert elicitation and to develop an approach to uncertainty/sensitivity analysis. Another area for further work will be to refine the methods for display and use of results.

Updates of the methodology will be based on insights of the work done on Demo Study and on the new study to be carried out in 2007-2008. It will be very important to further disseminate the results and to promote the use of methodology among potential users mainly among designers of advanced nuclear energy systems. By so doing the PR and PP concepts will be considered since the early design phases by accomplishing a PR and PP robustness by design.

## Acknowledgements

The full list of the PR&PP working group members is herewith reported together with the list of the other contributors. In addition all those who provided input and comments for the advancement of the methodology and on the case study reports are thankfully acknowledged.

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# Achieving a Global Cleanout of Highly-Enriched Uranium in all Non-Explosive Nuclear Applications before 2020

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**Abstract:**

*Highly-enriched uranium (HEU) is of particular proliferation concern due to the feasibility of constructing a crude nuclear explosive device from the material – making it the most likely choice for a terrorist group or non-state actors. In this paper, the 'HEU cleanout' concept and all existing categories of HEU-fuelled facilities have been systemized and analyzed to identify impediments and opportunities for achieving a HEU clean-out before 2020. The largest areas of HEU consumption, which also containing the largest number of HEU-fuelled facilities, are in the military sphere, in particular naval propulsion. The HEU-fuelled universe in 2006 represented the consumption of approximately 10 metric tons of HEU in more than 300 reactors covering both the military and the civilian sphere. The main conclusion is that if the current effort is not accelerated, a majority of the current HEU-fuelled facilities and processes will still be in operation more than 10 years from now. Most small and intermediate sized civilian reactors can be converted without significant technical impediments or loss in reactor performance. The justification for HEU-fuelled high-flux facilities should be evaluated more closely while the concept of shared facilities or centres of excellence is further developed, and increased emphasis should be put on decommissioning. New power reactor concepts can be developed without the use of HEU. There is a need for further consideration of the conversion of targets for large Mo-99 production facilities. The main impediment to a global cleanout is a lack of commitment in countries with advanced nuclear fuel technology infrastructure to decommission or convert HEU-fuelled facilities and fund the relevant international activities, such as the G-8 Global Partnership and Global Threat Reduction Initiative.*

**Keywords:** HEU; conversion; minimization; clean-out

## 1. Introduction

Highly enriched uranium (HEU) is an often used ingredient in most nuclear explosive and various significant non-explosive applications. HEU is also considered more apt than plutonium for use in an improvised nuclear device (IND). Thus, a recognized objective internationally has been to minimize the use of HEU and reduce the number of locations with HEU. Yet, nearing the 30 year anniversary of this objective, the number of HEU-fuelled installations in operation remains high, HEU is still being used in large quantities, and significant quantities of HEU are still to be found in a large number of unsecured locations worldwide. This paper analyzes the technical content of a possible HEU clean-out, and, on this basis, suggests future measures for achieving a real clean-out in all nuclear non-explosive applications. The different location and types of HEU-fuelled facilities discussed include a) propulsion reactors, b) power-related reactors, c) research reactors (steady-state reactors, critical and subcritical assemblies and pulsed reactors). For each of these areas, the present use of HEU is discussed together with the status of current HEU conversion and minimization efforts and the prospects for accelerated HEU phase-out. On the basis of the available technical data and the number of currently operating reactors, annual HEU consumptions are assessed and predictions through 2020 given.

### 1.1. Scope and limitations

The concept 'HEU clean-out' has been applied in various ways, either covering only civilian facilities or all types of HEU with emphasis on weapons material as this constitutes the largest single category of HEU. In this paper, the term 'HEU-clean-out' comprises all HEU-fuelled facilities (i.e. non-explosive applications). In addition, one has to assume that large amounts of HEU reside at various fuel fabrication, fuel reprocessing and storage facilities; however, due to a lack of available date these are not included in the scope of this paper. Any commitment to clean out HEU. however, will clearly have to deal with HEU fuels and HEU in other forms, at all facility types. Similarly, spent fuel at HEU-fuelled facilities is not discussed in any depth; it is only briefly considered together with each of the categories of facilities in chapter 2.

This study is criteria-based; it uses enrichment level and U-235 consumption in the assessment of priority measures. The figures for U-235 consumption include remaining quantities of U-235 in the spent fuel. The risk associated with HEU fuel may vary considerably with regard to these properties, e.g. irradiation time or burn-up, and fuel composition. The two former attributes has been discussed recently [1], with the conclusion that the 20% limit for fuel enrichment for converted facilities should be upheld as a sharp limit. Thus, in this paper all applications using fuel enriched above 20% will be discussed. Regarding the various types of facilities, there are obvious differences in risk in relation to the annual consumption of HEU vs. all facilities at the site, the various types of applications and the amount of fresh and spent material moved in and out of the core and stored at the site. As there in principle is no difference in risk with respect to the fact of whether the material is being used for civilian or military purposes, all facilities have been included. The argument that military facilities have better physical protection deserves to be tested on a case-by-case basis; it is noteworthy that the U.S. Department of Energy laboratory at Sandia decided to shut down use of one HEU reactor in lieu of expending millions of extra dollars in new post-September 11, 2001 security measures. Australia has upgraded security requirements in the civilian sphere, also leading to a choice against HEU use.

## 2. The HEU-fuelled Universe

The present concern for HEU in research facilities can be traced back to the International Fuel Cycle Evaluation (INFCE), launched in 1977 and completed three years later. It was realized then that the widespread use of HEU in different types of applications posed significant proliferation risks. The INFCE study led representatives from 59 states to agree that: "The trade in and widespread use of highly enriched uranium and the production of fissile materials constitute proliferation risks with which INFCE is concerned." [2]. The INFCE study recognized that there were over 140 HEU-fuelled research reactors with significant power-output (between 10 kWt and 250 MWt) in operation in more than 35 countries, each year producing in excess of 1700 MW and consuming more than 1200 kg U-235. As then, the current HEU-fuelled universe includes far more facilities than just research reactors; propulsion reactors, breeder reactors and isotope producing facilities play a significant role in the use of HEU in fresh fuel, as in-core or target material or spent fuel.

### 2.1. Research reactors

Most nuclear research facilities, in many cases small and compact, yet versatile, allows for wide-ranging and multiple use. A research reactor is a reactor used as a research tool for basic or applied research or for training: *"a nuclear reactor used mainly for the generation and utilization of neutron flux (…) for research and other purposes."* [3]. Facilities commonly known as critical and subcritical assemblies, pulsed and fast burst reactors are included in this definition. Research reactors come in variants less than ~W to well above ~MW. The generated fission heat is normally not used for electricity production or other purposes. As the complete status and existence of the different types and location of the existing HEU-fuelled research reactors are not well known, a summary of the information available has been listed in Appendix 1. As seen in Table 2.1, still almost 140 research reactors facilities remain in operation that fully or partly use HEU, with a nominal power of 1100 MW in total. In this assessment, all types of reactors are included, also military reactors. However, as seen in

Appendix 1, there are no military facilities registered among the high-flux facilities having nominal power above 1 MW, except for the isotope and Pu-production reactors discussed in chapter 2.3.

The pulsed reactors come in all different types and sizes, some also being steady state reactors operating in pulsed mode. The nature and number of the pulsed modulus vary with reactor size and external need; from a limited number pulses a day or pulses with a regular frequency using for example rotors in the core. Thus, the actual number of fissions – fuel consumption – is low; most facilities have lifetime cores. The existing pulsed reactors are found in Russia (at least 8 civilian and military facilities), Uzbekistan, the United Kingdom and the United States (one military facility in Sandia). An example regarding the amount of HEU installed in a pulsed facility is the Russian BIGR pulsed reactor at the Institute of Experimental Physics at Sarov (VNIIEF); containing 833 kg of 90% HEU as part of the core [4]. In critical assemblies chain reactions may be sustained by means of apt core configuration and appropriate controls. Sub-critical assemblies use the same reactor designs and configuration, but they are incapable of sustaining a chain reaction, either due to the assembly geometry or due to a limited amount of fissile material in the core. Usually such facilities also have life-time cores; examples of fissile material inventory are 56 kg in the Japanese FCA Tokai assembly (enrichment 20-93%) and 39 kg in the US ATRC assembly (enrichment 93%).

| | | Russia & NIS | China | Europe | US | Other | Total |
|---|---|---|---|---|---|---|---|
| Critical assemblies | | 34 | 2 | 6 | 1 | 2 | 45 |
| Pulsed reactors | | 14 | 0 | 1 | 3 | - | 18 |
| Steady-state reactors | 0,03 – 0,25 MW | 1 | 4 | 6 | 3 | 12 | 26 |
| | 0,25 – 1 MW | 1 | - | - | 1 | 2 | 4 |
| | 1 – 2 MW | - | - | 1 | 4 | 2 | 7 |
| | 2 – 10 MW | 7 | 1 | 1 | 2 | 2 | 13 |
| | 10 – 250 MW | 9 | 1 | 8 | 4 | 2 | 24 |
| Total | | 66 | 8 | 23 | 18 | 22 | 137 |

**Table 2.1:** HEU-fuelled research reactors in operation – 2007

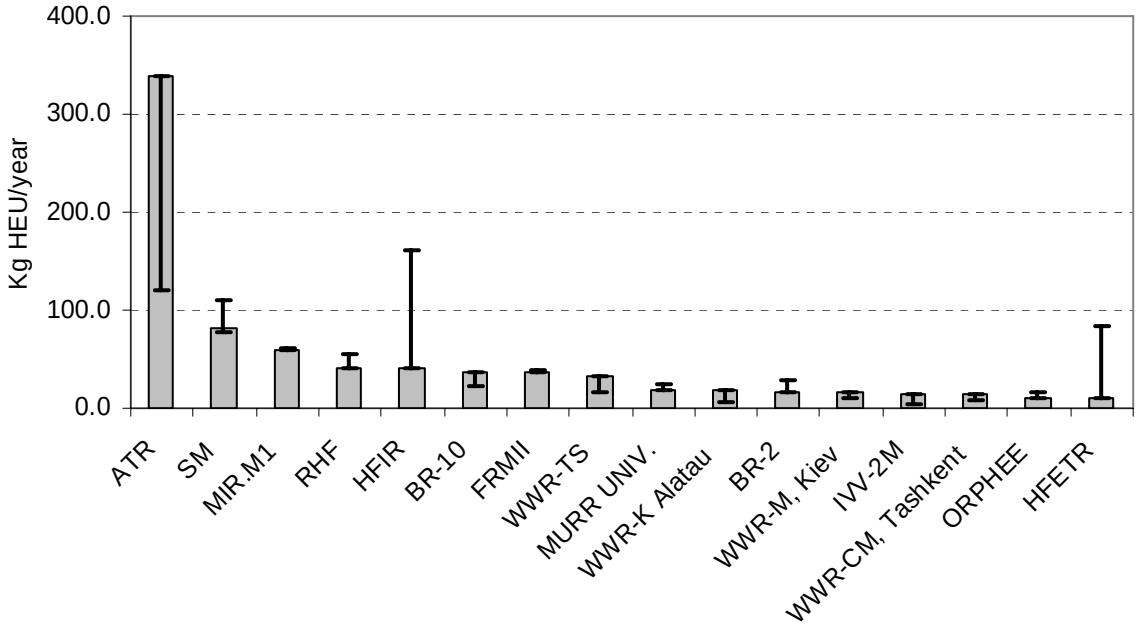


**Figure 2.1:** HEU Consumption in research reactors – 2006

The HEU consumption varies considerably for the steady-state reactors in Table 2.1. To a certain extent, information on individual facilities has been removed from the public domain, most notably from

the IAEA Research Reactor Database (RRDB). As the basis for Figure 2.1, the data in RRDB on nominal power, average burn-up and availability has been applied to calculate the annual consumption of U-235. The error bars indicate the span (min./ max.) of consumption estimates for each reactor given in various sources. [5] [6] [7] [8] Sources of uncertainty come from lack of relevant information in the IAEA RRDB and/ or because of data on availability do not correspond with the actual situation. For example, the ATR reactor is said in the current version of the RRDB (last updated in 2001 by US competent authorities) to operate 24/7 52 weeks each year, which leads to unrealistic results for its fuel consumption. It has also been claimed that the figures for availability reported to the IAEA are larger than the actual situation in order to justify continued operation. For 2006, the facilities discussed in Figure 2.1 constitute, using the RRDB values for all facilities, 94% of the total consumption of HEU in all research reactors.

Another area related to research reactors where HEU is being consumed, and in increasing quantities as seen in Figure 2.2, is the production of Mo-99 – the basis for producing Tc-99, the most widely used radioactive isotope in nuclear diagnostics. Mo-99-production may involve HEU or LEU as fissionable target material, though there are also other technologies available, such as solution reactors with fuel permanently dissolved in a liquid, and neutron activation techniques using Mo-98. When using fission, Mo-99 is produced from the reaction U235(n, f)Mo99, with 6,1% yield. The product is often assessed in 6-day Ci; the amount of Mo-99 remaining 6 days after being transported to medical companies worldwide. The main production centres using HEU-based technology, covering currently almost 90% of the Mo-99 used for medical purposes [9], are to the following four installations: the BR-2 reactor, also using HEU in its driver fuel, the Dutch High Flux Reactor (HFR), NRU at Chalk River in Canada and the South African SAFARI reactor. In Figure 2.2, figures for the yield when producing 6-day Ci from HEU using the Cintichem process have been used to assess the overall amount of U-235 – as HEU (36% enriched in the case of South Africa, 90% for the others) – consumed in this process. The calculations are based on a 10% efficiency loss when extracting the Mo-99 from the targets after irradiation and an annual 1% gain through technology improvements. The main feature is the growth; in figure 2.4 the annual growth has been assumed to be 10%. While this technology only a few years ago involved tens of kilograms of HEU, the annual consumption may soon reach 100 kg of HEU. At the same time, a significant increase in the use of non-HEU based technology has to take place to compensate for the large increase in the global Mo-99 market.

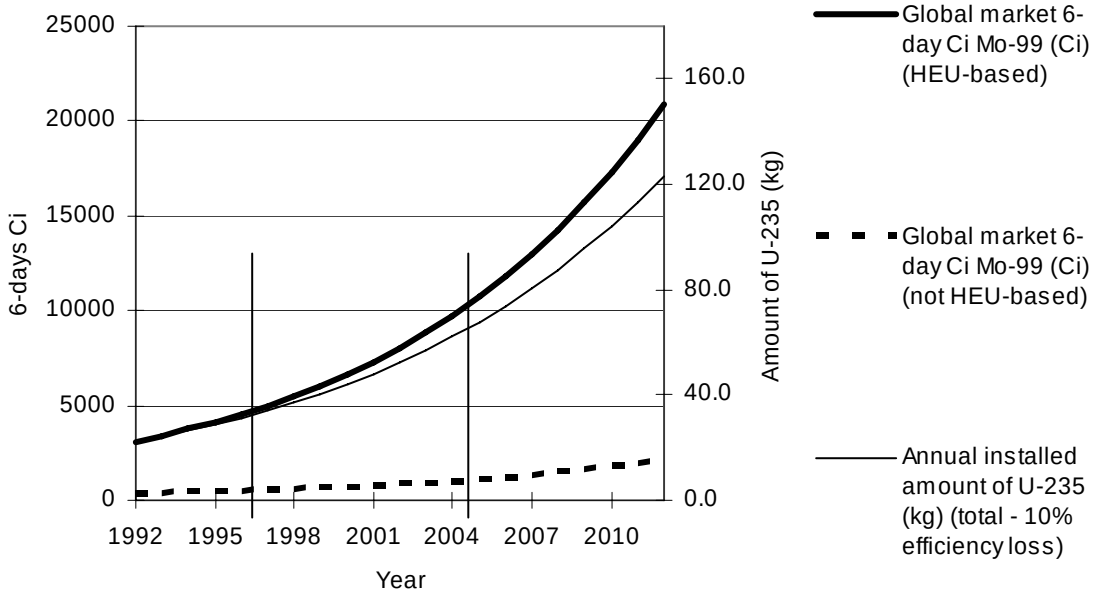


**Figure 2.2:** Projected Mo-99 production market size and U-235 consumption in research reactors – based on 10% annual growth and market size figures for 1996 and 2005

Two HEU-fuelled facilities probably still involved in the production of other types of isotopes are also the Russian light-water reactors *Ruslan* and *Ludmila*. As these are military facilities part of the Mayak complex, very little about their operational status, power level and fuel is publicly available. However, generally considered to be about 1000 MWt and using HEU to spike-up the flux in the isotope-producing regions, the U-235 consumption has been assessed to 230 kg, or 750 kg HEU annually. [10]

## 2.2. Propulsion reactors

In nuclear propulsion – marine vessels such as cruisers, aircraft carriers and submarines as well as space reactors – HEU has traditionally been the preferred reactor fuel. A somewhat exotic HEU application is found in Northwest-Russia as fuel for the nuclear-propelled icebreaker fleet, also used as test beds for the development of Russian nuclear submarines. [11] [12] As seen in Table 2.2, the HEU-fuelled nuclear propulsion inventory presently includes 152 reactors in three countries; France has seemingly left this exclusive club, as all current naval reactors reportedly are LEU fuelled except possibly the reactor in *L'Inflexible* which will be decommissioned in a few years (it has not been included in Table 2.2 and Figure 2.3). There are no indications which direction China is taking their nuclear propulsion program, until now the country is, as earlier pointed out, believed to have their propulsion reactors operating on LEU cores. [13] Earlier, also other countries such as Argentina and Brazil have aimed for a nuclear propulsion capability as part of their defence-related installations, as India probably is doing at the moment.

| | Russia | | US | UK |
|---|---|---|---|---|
| | Civilian | Military | Military | Military |
| Nominal reactor power < 250 MW | 9 | 30 | 79 | 14 |
| Nominal reactor power > 250 MW | - | 2 | 18 | - |

**Table 2.2:** HEU-fuelled Propulsion Reactors in Operation – 2006

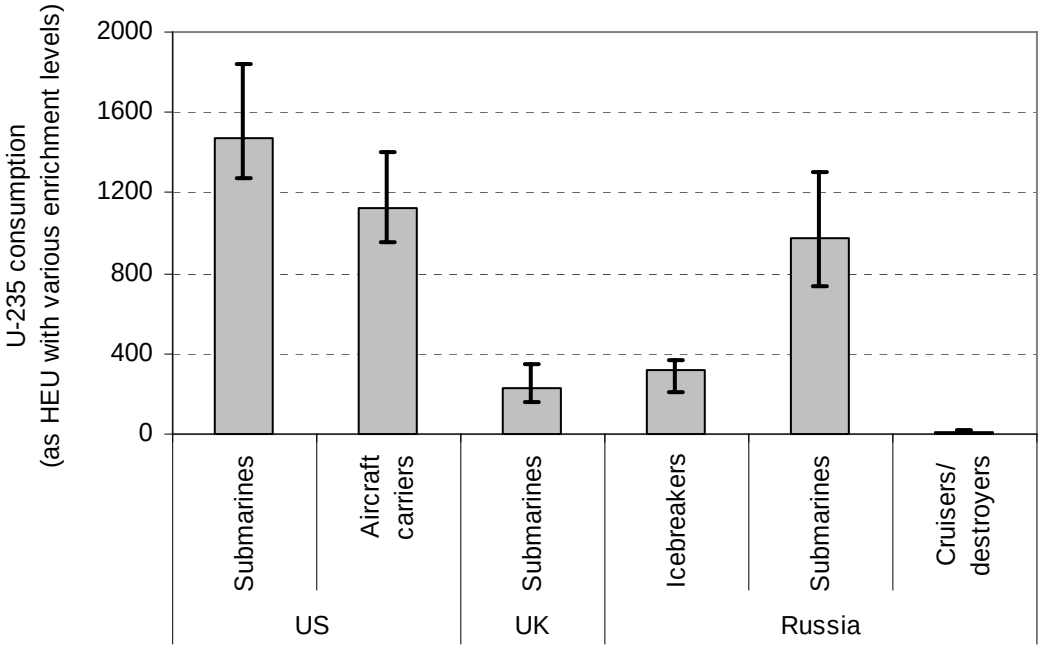


**Figure 2.3:** HEU consumption in naval propulsion facilities 2006

The present consumption (2006) of U-235 in fuel initially enriched above 20% for the world's civilian and military nuclear-propelled fleet has been calculated in Figure 2.3. The basis is an assessment of a) the annual consumption of cores for each generation of vessels (one to four) which generally correspond to the reactor generations, and b) a core inventory. The core inventory is traditionally not publicly available, thus, these figures are in most cases based on available data on core-life and burn-up, and assumed operational activity.

As the United States started early to use weapons-grade HEU – 97,3% enriched – and currently has by far the largest nuclear fleet the United States naval facilities dominate the picture. A major component is the carriers with several reactors with high nominal power and large inventories of fuel. In total, the United States has 93 reactors currently in operation, consuming about 2600 kg U-235, out of which 40 % is consumed in the operation of the aircraft carriers as seen in Figure 2.3. The UK Navy had in 2006 13 naval reactors consuming approximately 230 kg U-235, initially enriched to the same level as for the United States naval reactors. Russia is in a peculiar situation as probably only civilian vessels use 90% enriched HEU, while the submarines and surface vessels probably use medium-level HEU fuel (20-45%). A possible hypothesis is that the Soviet Union, out of concern for the overall operational reliability (redundancy) historically preferred to equip their vessels with two reactors, hence having no need for optimizing fuel efficiency and packing. [11] The most recent 3rd generation vessels have one reactor (OK 650 B/ VM-5). There is, however, little data available to indicate the initial core load, operational capacity or any other information on if and how the reduction in reactor power has been compensated for when using a single reactor configuration. Due to low operational activity and only a few commissioned vessels after 1990, probably less than 20 cores have been consumed by this reactor generation. For 2006, the HEU consumption figures are 0,3 tons (90% enriched) for the civilian fleet. The figure for the Russian submarine fleet is somewhat less than 1 ton U-235 (20-45%).

Regarding space propulsion, both the Soviet Union and United States both have used HEU-fuelled reactors. [14] No such facilities are currently in operation. According to the five year plan for the United States DOE Nuclear Criticality Safety Program, NASA continues to be interested in benchmark experiments for their proposed space reactor to power the Jupiter Icy Moons Orbiter [15]. As the concept of using HEU in space missions has some appealing features with respect to the weight and endurance of the power facility, it has been suggested that the main problem today is rather to have countries completely renounce the option than that the technology is reintroduced any time soon.

## 2.3. Power-generating reactors

While today's commercial power reactors use low-enriched UO2 or mixed oxide (MOX) fuels, the question of whether HEU has any role in the development of new reactor systems, particularly fast reactors, has been raised on several occasions. Several countries have in the past built prototype fast breeder reactors that have used or use HEU fuels. Today, as given in Table 2.3, only one breeder reactor fuelled with HEU is currently in operation (the Russian BN-600). In addition to these facilities, there are a number of fast reactor research facilities, such as the Russian *BR-10* and *BOR-60* facilities, and the neutron sources *RSV Tapiro* and *YAYOI* which in this paper are included in chapter 2.1.

| Country | Facility | Reactor type, nominal power (MW) | U-235 consumption (kg.) (% initial enrichment) | Comments |
|---|---|---|---|---|
| Russia | Seversk, AD-4 | Pu-production, (approx. 2000 MW) | 200 (90%) | First criticality 1965, projected shut-down 2008 |
| Russia | Seversk, AD-5 | Pu-production, (approx. 2000 MW) | 200 (90%) | First criticality 1967, projected shut-down 2008 |
| Russia | Zheleznogorsk | Pu-production, (approx. 2000 MW) | 200 (90%) | First criticality 1964, projected shut-down 2011 |
| Russia | BN-600 | Fast breeder, 600 MWt | 4000 (HEU, 20-25%) | First criticality 1973 |

**Table 2.3:** HEU-fuelled Breeder and Pu-production reactors in Operation 2007 and U-235 Consumption

The operational HEU-fuelled breeder and Pu-production facilities have a very different profile in their HEU consumption; while only part of the fuel for the Russian BN-600 is enriched just above 20%, the Pu-production facilities use 90% enriched HEU. Regarding the Russian production reactors, HEU fuel has been claimed to account for 10% of the power generation according to [15], with 30% burn-up and a nominal effect on 2000 MWt, each facility uses about 200 kg HEU (90%) annually. Russia has now agreed to close the Pu-production facilities in Seversk and Zhelenogorsk in 2008 and 2011, respectively.

Limiting new reactor designs, including breeders, to LEU will place few if any limitations on developing future advanced power reactor designs. [10] None of the designs under development today, either through the IAEA's International Project on Innovative Nuclear Reactors and Fuel Programs (INPRO) or through the Generation IV International Forum (GIF) program, call for the use of HEU. The Russian program, like others worldwide, does not envisage the use of HEU in new generations of fast breeder reactors [16]. However, the testing of fuel for future fast reactors does, at present, involve the use of HEU in critical assemblies, for example at France's Masurca facility at Cadarache. Cadarache does not have enough plutonium to undertake these experiments without HEU at this time. It is noteworthy, however, that "enrichment higher than 30-35% does not seem to be needed to mock-up conceivable core design as proposed today," and also that there are no new needs for HEU fuels for future generation fast reactors. [16]

While there are only a few fast HEU-fuelled reactors in operation, the decommissioned fast facility BN-350 in Kazakhstan has caused considerable concern the last years as large amounts (~of HEU enriched to 26%) has been stored. In October 2005, the removal of this material was initiated by the government of Kazakhstan in close cooperation with NTI [17].

## 3. The HEU Minimization Agenda – Facility Conversion and Shutdown

The two main avenues for minimization of HEU-fuelled facilities are *decommissioning* and *conversion*. When considering the progress in HEU minimization since the INFCE-study was completed, these two processes have, as seen in Figure 3.1 resulted in an equal decrease of HEU consumption regarding research reactors. Thus, a main issue for the future is to design the global research reactor sector in a way that makes it possible to decommission outdated, not justified and under-utilized HEU-fuelled reactors of various types and sizes. The sudden increase in 2004 was due to the commissioning of the German FRM-II reactor. As the limited need for using HEU in breeder reactors already has been discussed, these facilities will not be further addressed here. When considering the development described in Figure 3.1 and 3.2, there is a need to consider the future potential for conversion vs. decommissioning.

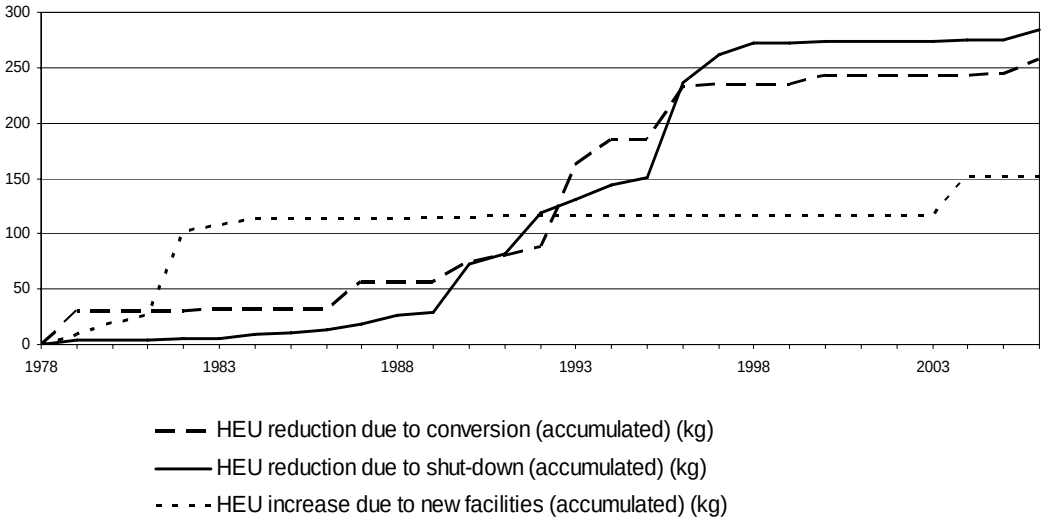

**Figure 3.1:** Changes in U-235 Consumption for Research Reactors – Conversion vs. Decommissioning 1978 – 2006

Large efforts have been ongoing since before the INFCE-study was finished to convert existing facilities. Regarding the civilian sector, the United States initiated the Reduced Enrichment for Research and Test Reactors (RERTR) program in 1978 to develop the technical means to convert HEU fuelled research reactors to the use of LEU and assist in the conversion process. A corresponding effort was initiated in the Soviet Union aiming at reducing the enrichment levels in foreign supplies to 36%. The Global Threat Reduction Initiative (GTRI) was launched in May 2004 in cooperation with the International Atomic Energy Agency (IAEA), but is operated out of the United States Department of Energy's (DOE) National Nuclear Security Administration (NNSA) with strong support from a number of countries. Today the GTRI includes all United States programs to protect and also assist in the removal of vulnerable material, among them the RERTR program. These programs are the initiative responsible for the projected decrease in HEU consumption as a result of conversion as described in Figure 3.1.

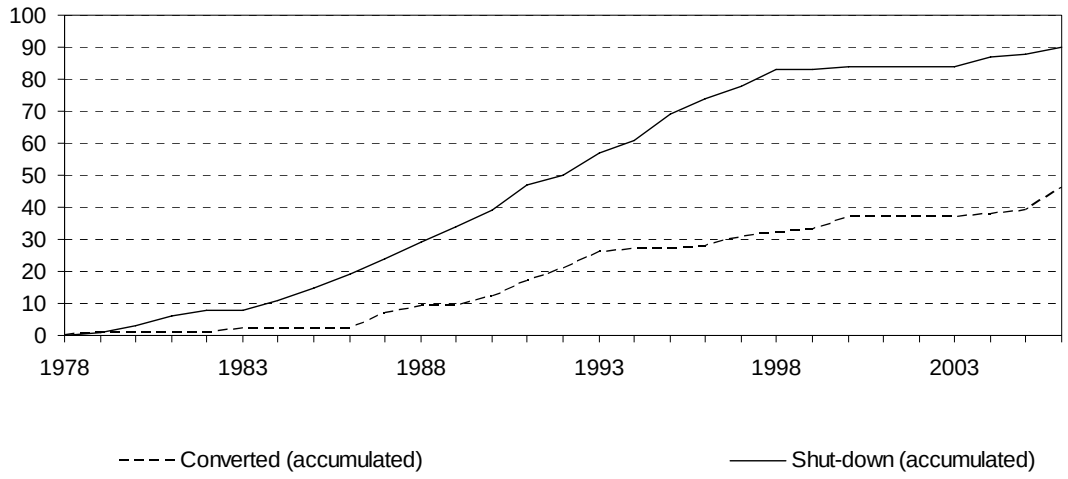


**Figure 3.2:** Changes in Number of HEU-fuelled Facilities – Conversion vs. Decommissioning 1978 - 2006

The decommissioning option, despite its successes as described above, has no international programs to support it, and has to be approached by addressing the possible incentives and the individual reactors. A recent initiative less concerned with conversion and more with minimization and clean-out was taken by the Norwegian government at the NPT Review Conference in 2005 [18]. In June 2006 an international symposium was held in Oslo with the objective of establishing international consensus on relevant technical issues related to HEU minimization, and for reaching agreement on the way forward for a concerted international effort within the framework of existing international agreements, organizations and assistance programs. [19] The results of the international conversion programs described above was taken note of; e.g. that most small and intermediate sized reactors can be converted without significant technical impediments or loss in reactor performance, and that new power reactor concepts can be developed without the use of HEU. Regarding HEU-fuelled high-flux facilities, it was agreed that the justification of those should be evaluated closely and in the view of further development of the concept of shared facilities or centres of excellence.

The GTRI has assessed the individual reactors in the RERTR – perspective as seen in Figure 3.3; the possibility for conversion, and has concluded that 49 reactors may be converted with existing fuels, 26 need new high-density fuels and 54 facilities are outside the scope of the GTRI program and have therefore not been assessed in this context. Before assessing the different categories in Figure 3.3, additional assumptions may be added as a basis for establishing the future agenda for HEU minimization: There is no reason for having HEU in low-power reactors below 1 MW, as the main purposes of these reactors are education and training, testing of instrumentation and, if suitable, neutron radiography. However, most of these facilities do have life-time cores and are therefore of less

concern when considering some of the criteria mentioned in chapter 1.1. Thus these facilities do not constitute an independent need for a conversion program. Since the Pu-production reactors have a well-defined shut-down date, those will not be further discussed below.

### 3.1 Research reactors

Out of the facilities listed in Figure 2.1, the WWR-reactors and HFETR are examples of reactors possible to convert with existing fuels, while the conversion of the other facilities is dependent on the development of new fuels (see Figure 3.3). The development of high-density fuels was restarted in the second half of the 1990-ties as the funding of the RERTR increased substantially. The main avenues chosen then were to continue the effort on silicide fuel, which had shown promising results with densities up to approximately 6 gU/ $cm^3$, and other dispersion fuel types containing U-Mo and U-Zr-Nb alloys. The Russian program insisted on focusing on uranium-oxide cermet fuel, which turned out to be a failure, partly due to the fact that Russian fuel fabricators did not have the ability produce the fuel at a reasonable cost. The most advanced LEU fuel available today is a dispersion fuel – $U_3Si_2Al$ with density 4.8 Ug/$cm^3$. Today this fuel type has been in use for over a decade.

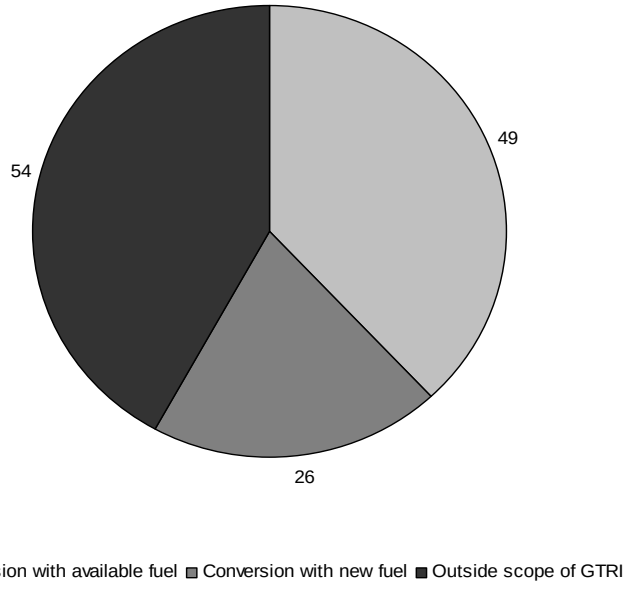


**Figure 3.3:** GTRI categories of HEU – fuelled research reactors - 2007

The other candidate high density fuel is a UMo alloy, often referred to as monolithic fuel. For some research reactors, for example the large HFIR research reactor in the US, U-Mo monolithic fuel with a density up to 16 g/$cm^3$ is needed for conversion to LEU. The expectations for a break-through have been large; in 2000 U-Mo 6 Ug/$cm^3$ dispersion fuel was planned to be qualified by the end of 2003 and 8–9 Ug/$cm^3$ by the end of 2005. In 2003, failed progress in the development of high-density fuel led to an initiative to extend the May 2006 deadline for the acceptance of spent fuel from HEU-fuelled facilities. Several ways to correct U-Mo fuel performance problems were subsequently investigated, including changes to fuel and matrix chemistry, replacement of the aluminium matrix with magnesium, and elimination of the matrix altogether. Two irradiation experiments to test the improved U-Mo fuels have not revealed fuel performance issues for most of the fuels, excepting two fuel specimens produced by transient liquid phase bonding [26]. The objective now is to complete the development and qualification of very-high density fuels by 2010 and the actual conversion by 2014 for most facilities. However, even if this fuel becomes available, there exist at least one reactor which cannot be converted to LEU without changes in fuel geometry; the German FRM-II reactor.

Even if accepting the current schedule for conversion of the current high-flux facilities, a large number of facilities will remain HEU-fuelled beyond 2014. In Figure 3.4, all projected conversions –primarily all US HEU-fuelled civilian facilities – are included, in addition to published decommissioning dates. A radical shift in HEU consumption in 2014 is noted; however, for most existing HEU-fuelled facilities no conversion or decommissioning date has been set. The main reason is that 1) a large part of these facilities has not been part of any minimisation program until recently, such as non-steady-state reactors and critical assemblies, 2) they are said to be outside the scope of these programs, being for example military facilities – as seen in Figure 3.3.

At the moment there are some discrepancies between the scope of the GTRI and the RERTR program. The latter program currently includes 12 critical assemblies on their list of candidates for conversion, however, no rational for this decision, or the lack of decision regarding any other of these life-time facilities given in Appendix 1, has been registered. Therefore, a priority task should be to assess the need for these facilities and create a new initiative within the GTRI for addressing all facilities of these types as soon as possible in the context of HEU minimization, and then also include decommissioning as an option. There should no need for using HEU in pulsed reactors, critical and subcritical assemblies except in very special cases. These facilities are installations which may be replaced by computer modelling; probably only a few facilities would cover the need for being able to physically building and adjusting a physical mock-up of critical material. Most HEU-fuelled critical facilities are no longer needed because neutronic codes for standard reactor types are well tested, and computers are now fast enough to make detailed virtual simulations of the reactor in question. However, conversion is not an option as the various options for simulating criticality disappear with different configurations or fuel materials.

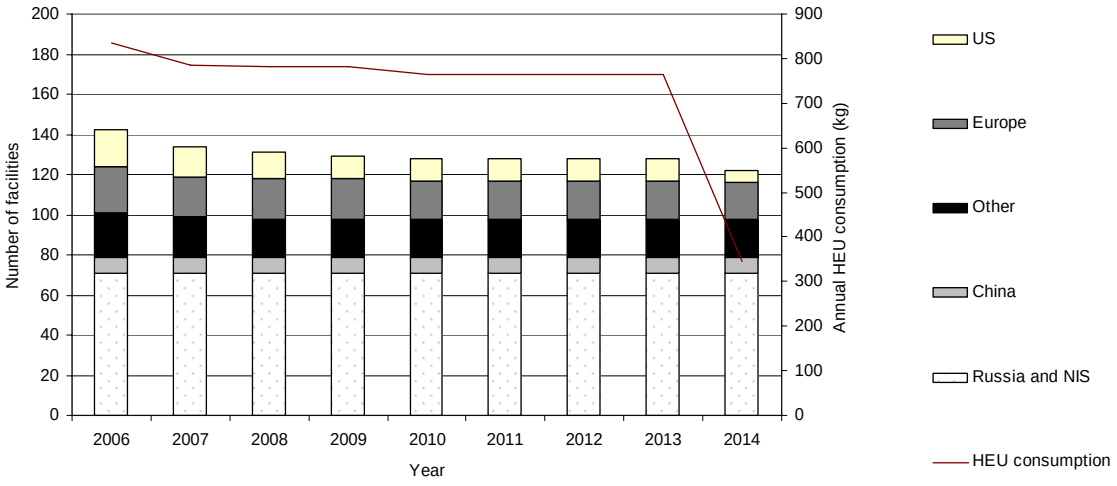


**Figure 3.4:** Projected Development Regarding the Conversion of Research Facilities 2006 - 2015

When the facility of which the critical assembly is a mock-up has been decommissioned or converted, the assembly loses its main justification. In addition, the research focus of nuclear engineering has changed; the development and testing of new reactor and fuel materials and power plants designs solutions has given place to standardization, exit strategies and aging considerations, thus, the small innovative facilities with exotic fuels are less needed. The increasing concern for safety and security, in particular when considering HEU-fuelled facilities, also for dismantling and decommissioning activities, makes it even more difficult to sustain the continued operation of the HEU-fuelled facilities. However, currently decommissioning cost may even be a driving force for continued operation as no governmental funds have been set aside for this purpose.

The IAEA has dedicated resources to issue guidance documents for the strategic planning for research reactors in order to establish a strategic plan which *"provides the rationale for the future for the facility"*. [20] The IAEA has also made clear that it will only *"support requests for new facilities or equipment for research reactor utilization if they are accompanied by a strategic implementation plan clearly demonstrating that the items requested are necessary to achieve the plan."* [21] Today more

proliferation resistant solutions should be considered in addition to the possibility of having shared facilities or centres of excellence. The US has now constructed an alternative source of high flux, the Spallation Neutron Source, which will be their *"frontier research facility for neutron scattering to analyze structure of matters and materials.*" [22] Such projects are in the multibillion-dollar class and thus beyond the scope of most states or even regions. On this basis, there is an obvious need for developing arrangements for sharing access to new high-flux facilities based on alternative technology. Another proposal is to consider how to get the most optimal results of the available experimental set-up – without using HEU – thus, how to develop LEU-fuelled high-flux facilities and or improve the experimental apparatus. [23]

The large unknown in this scenario is if the decommissioning effort will speed up in the years to come. When considering the age of the facilities in the two categories of facilities targeted for conversion in the years ahead (see Figure 3.3) – as seen in Figure 3.5 and 3.6, the majority of those for which the new fuels are being developed will on average soon exceed 40 years of operation. There is an obvious need for regions, such as the EU, or individual countries, to establish schedules for decommissioning or conversion. Regarding the former, there is – at the strategic level of research reactor capabilities – a growing concern among operators about how to justify their existence in a shrinking market with lower governmental subsidies.


**Figure 3.5:** Year of first criticality - research reactors which can be converted with existing fuels

As isotope demand is increasing 5-10% pr. year, the argument for having one new high –flux reactor in Europe for isotope production online by 2010-15, and another by 2015-2020 has been made. However, the emerging Mo-99 production activities in Argentina, Romania, Indonesia, Australia and US, at the University of Missouri, have shown that there are no fundamental a priori technical impediments for establishing new production capacities producing $^{99}$Mo based on LEU, and that the emerging norm is to use LEU technology. The current lack of progress regarding conversion has led the US Congress to engage the US National Academy of Sciences (NAS) to assess ways and means to provide new options for how to alleviate the current dependence on HEU in the production of Mo-99 for domestic use in the US. There are currently no specific plans with the four main producers to convert to LEU targets, however, recent NAS study meetings suggest that these producers have begun to accept the idea that they will eventually have to convert. One of the main objectives for Mo-99 production should be that all future production will be established on the basis of LEU technology. This is the focus of an IAEA Coordinated Research Program which seems to be succeeding in getting potential producers globally together for an efficient dissemination of the available knowledge on isotope production processes, target fabrication and waste handling.

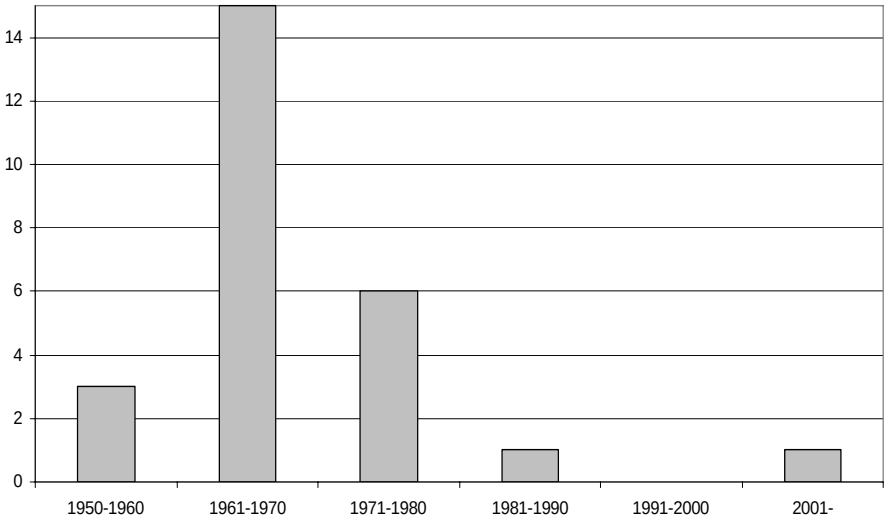


**Figure 3.6:** Year of first criticality - research reactors which require new fuels for conversion

## 3.2 Military Naval Reactors

The military naval reactors are the area with the most reactors and the largest amount of material in use in 2006. When projecting the use of HEU into the period between 2006 and 2020, as seen in Figure 3.4, one notes that there are no plans to reduce the use of HEU in the United States, United Kingdom or Russia, while France is about to phase out their reliance on HEU in their military vessels. France made a decision in the 1970s to run their new Rubis submarine on LEU fuel. The Rubis-class is the smallest nuclear submarine ever built with a displacement of 2500 tons. The US, Russia, and the UK have shown no interest in similar initiatives with respect to their navies or, in the case of Russia, its icebreaker fleet. The United States' long-term plan seems to be to introduce one Virginia-class submarine every year until 2020, while reducing the number of Los Angeles-class by one every year. The influence on annual consumption is small as seen in Figure 3.1; the overall level of 2,6 metric tons will remain constant. The replacement of the *Vanguard*-class in UK will have effect only at a point in time well beyond 2020. The United States has reserved 180 tons of HEU for naval and other reactor fuel. The UK has stated that none of their HEU stocks will be declared excess as the material has been set aside for their submarine program. Russia is assumed to have reserved 100 tons of HEU for naval reactors. The main arguments for maintaining high enrichment levels are compact cores and high endurance, in the most recent vessels the reactor does not need to be refuelled as core life-time corresponds to the life time of the vessel itself.

The United States Navy was challenged in 1995 to assess the potential for conversion of the submarine cores to LEU, and the options of adjusting endurance and/or core volume were assessed. The main conclusion was that LEU-fuelled reactors would, among other elements, cause greater occupational radiation exposure, generate more waste and have considerable economic consequences. To preserve the longevity of the core, the core volume had to be increased threefold. Subsequently the pressure vessel, the reactor compartment and the size and the cost of the vessel itself would have to increase correspondingly; according to the assessment construction costs would increase "about 28% for aircraft carriers and 26% for submarines – about $1.1 billion pr. year" [24]. The assessment was made without any reference to the possible larger implications of continued operation on HEU. However, the most important inherent problem with the assessment was, due to the high level of secrecy surrounding the design and operation of these installations, the lack of details. The conclusion in one of the few open-source studies on conversion of nuclear-propelled reactors was that the dimensional increase is sufficiently small that it can easily be compensated for by the use of an integral reactor design as used in the French Rubis. [25] The UK is currently in the middle of making a decision on what to do when the operational life-time of the naval reactor in the Vanguard-class boats end in the mid 2020s.

The UK should include an assessment of the potential for using LEU when designing the replacement for the PWR-2 in the Vanguard-class. The potential content of an effort to consider the potential for conversion of nuclear submarines would be to 1) assess the French approach; rationale and relevance for other countries, 2) complete an independent assessment on a) fuel design, b) impact on the pressure vessel, reactor compartment and, subsequently, on the operational parameters. The United States probably has with the recent introduction of life-time cores a considerable advantage regarding optimization of operational costs and, possible, directly related properties such as diving depth and endurance, compared to for example Russian vessels. However, continued effort should be put into translating design modifications into operational properties in a transparent way to pave the way for an evaluation of the pros and cons of continued operation with HEU fuel. As long as the area of military propulsion is not addressed at all, this may affect support for the US-led effort in the civilian nuclear sector. The most promising area is the Russian icebreaker technology as more space is available and, due to intensive operations, the fact that these vessels are being refuelled quite frequently already (several vessels a year). A naval reactor using LEU has already been developed as a basis for the Russian floating power concept where a prototype is currently under construction, and one study has been completed showing the potential for converting these reactors to LEU using high-density fuel types. This study should have a follow-up with respect to other types of Russian-designed naval reactors. Regarding a future Chinese and Indian nuclear-propelled navy, the current UK deliberations are potentially extremely important, as they will be a signal to evaluate more closely the potential for LEU-fuelled submarines for the near future.

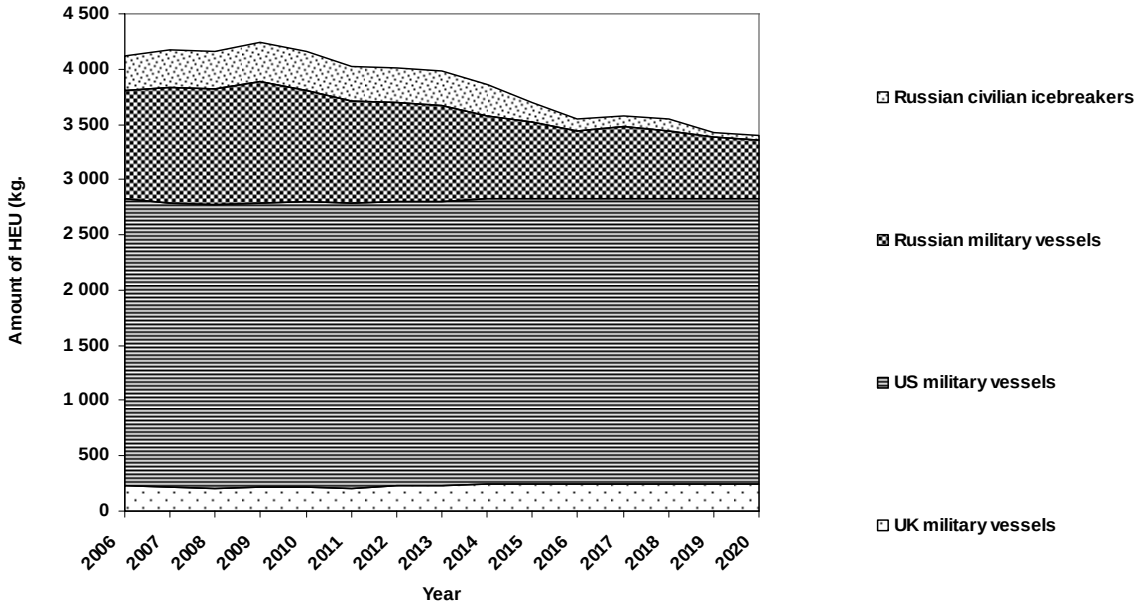


**Figure 3.7:** Projected HEU consumption in naval propulsion facilities 2006 - 2020

When considering the role of decommissioning HEU-fuelled propulsion reactors, Russia stands out – possibly involuntary – as a promising case. While HEU minimization seems not to be of particular importance to the Russian government in other areas, the decline of HEU consumption in the Russian Navy will continue as old vessels are taken out of service and the introduction rate of new vessels remains low. The question is whether Russia will aim to construct new military vessels with higher endurance – probably life-time cores – if not convinced otherwise. A credible scenario for the nuclear icebreakers is to include the commissioning of the brand new vessel, '50 Years of Victory', recently completing trials outside St. Petersburg, and another vessel in 2012. Given Russia's hopes to expand use of the Northern Sea Route to include freight shipments between Asia and Europe, the construction of additional icebreakers is not unlikely in the future. Russia should be encouraged to explore the use of LEU for these future vessels.

## 4. Conclusions and Recommendations

Today, there are over 300 civilian and military HEU-fuelled facilities consuming approximately 10 metric tons of HEU per year. A large part of this material remains as HEU after being used as fuel or target material. The main contributor is nuclear propulsion, in particular military naval vessels, and other various types of facilities in Russia and the United States. Isotope production and civilian research reactors in other countries also represent areas of considerable importance when aiming for an all-encompassing HEU clean-out in non-explosive applications. A considerable amount of effort and resources are being spent on international conversion efforts, most notably the GTRI initiative. The objective of GTRI is to convert all targeted civilian facilities by 2014. However, without a radical change in the international approach to the continued operation of HEU-fuelled facilities, in particular in Russia, a large number will continue to be in operation in 2020.

The current initiatives only address civilian facilities, and use fuel development and reactor conversion as their primary tool for achieving HEU minimization. The justification of HEU-fuelled high-flux facilities should be evaluated closely, in concert with the further development of the concept of shared facilities or centres of excellence, while increased emphasis should be put on decommissioning. New power reactor concepts can be developed without the use of HEU. There is a need for further impetus to be given to the conversion of targets for large Mo-99 production facilities; here the technical problems have been overcome, but conversion cannot move forward until governments have determined who will pay for that conversion and related tasks. The main impediment to a full HEU clean-out is the lack of commitment in countries with advanced nuclear fuel technology infrastructure to decommission or convert HEU-fuelled facilities and fund the relevant international activities, such as the GTRI and the G-8 Global Partnership. When considering the future prospects for reactor conversion, one must conclude that the the international community has failed to prioritize the need for reduced risk of diversion of HEU above the needs of individual institutes for HEU flux and related services, and that countries like the United States, Russia and the United Kingdom see no significant proliferation risk associated with the use of large quantities of HEU in the military sphere, despite the fact that this choice makes it far more difficult to persuade other countries to reduce HEU use in the civilian sphere.

In order to reduce the risks of HEU use as effectively as possible, enrichment level and HEU consumption should be the criteria used to assess the priority of various measures. Thus, the minimum possible enrichment level should be used for each necessary task, and quantities of HEU should also be held at a minimum, and in the minimum number of maximum-security sites. For example, if maintaining one facility in Europe to test new fast reactor fuels is necessary, then this facility (Cadarache's Masurca) should be shared by all concerned and use the minimum enrichment level (no more than 30%).

When considering past experiences, reactor decommissioning has been as important in reducing the number of HEU-fuelled facilities and HEU consumption as has conversion. A separate program for the decommissioning of HEU-fuelled facilities should be initiated, that provides other alternatives to the relevant sites instead of the continued operation of old and outdated facilities. At the present time, decommissioning is often not considered as a realistic option due to both to the costs of decommissioning itself, and the loss of employment for relevant employees without the funds to set up alternative work installations. This might be one element in the revitalization of the GTRI initiative: establishment of a program covering all types of facilities worldwide, including the facilitation of scientific work at shared "centres of excellence" and funding of decommissioning.

## 5. Acknowledgements

This paper is based on an effort with the International Panel on Fissile Materials (IPFM) (www.fissilematerials.org) to develop the technical basis for practical and achievable policy initiatives to secure, consolidate, and reduce stockpiles of highly enriched uranium and plutonium, in this regard a future report on HEU clean-out. The authors would therefore like to express our gratitude for the continued exchange of opinions and knowledge with the members of IPFM, in particular Dr. Matthew Bunn, Dr. Alexander Glaser and IPFM co-chair Frank von Hippel. Any mistakes, errors and inconsistencies are the sole responsibility of the authors.

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# Appendix 1: List of operational HEU-fuelled Research Reactors*

| Country | Name of facility | Type of facility | Civilian/ military | First criticality | Power level (MWt) |
|---|---|---|---|---|---|
| Argentina | RA-6 | Steady state | Civ. | 1982 | 0.5 |
| Belarus | YALINA-Booster | Critical assembly | Civ. | | |
| Belgium | BR-2 | Steady state | Civ. | 1961 | 100 |
| Canada | Slowpoke Alberta | Steady state | Civ. | 1977 | 0.02 |
| Canada | Slowpoke Saskatchewan | Steady state | Civ. | 1981 | 0.02 |
| Canada | Slowpoke-2 Halifax | Steady state | Civ. | 1976 | 0.02 |
| Chile | RECH-2 | Steady state | Civ. | 1989 | |
| China | HFETR Critical | Critical assembly | Civ. | 1979 | 0 |
| China | Zero Power Fast Reactor | Critical assembly | Civ. | 1970 | 0.00005 |
| China | HFETR | Steady state | Civ. | 1979 | 125 |
| China | IAE | Steady state | Civ. | 1984 | 0.027 |
| China | MJTR | Steady state | Civ. | 1991 | 5 |
| China | SD | Steady state | Civ. | 1989 | 0.033 |
| China | SH | Steady state | Civ. | 1991 | 0.03 |
| China | SZ | Steady state | Civ. | 1988 | 0.03 |
| Czech Republic | LVR-15 REZ | Steady state | Civ. | 1957 | 10 |
| Dem. P.R. of Korea | IRT-DPRK | Steady state | Civ. | 1965 | 8 |
| Former Yugoslavia | RB | Critical assembly | Civ. | 1959 | |
| France | Eole | Critical assembly | Civ. | 1965 | |
| France | MASURCA | Critical assembly | Civ. | 1966 | 0.003 |
| France | MINERVE | Critical assembly | Civ. | 1959 | 0.0001 |
| France | Pile Azur | Critical assembly | Mil. | 1962 | 0.0001 |
| France | ORPHEE | Steady state | Civ. | 1980 | 14 |
| France | RHF | Steady state | Civ. | 1971 | 58.3 |
| France | SILENE | Steady state | Mil. | 1974 | 0.001 |
| Germany | FRMII | Steady state | Civ. | 2004 | 20 |
| Ghana | GHARR-1 | Steady state | Civ. | 1994 | 0.03 |
| Hungary | BUDAPEST RES. REACTOR | Steady state | Civ. | 1959 | 10 |
| India | APSARA | Steady state | Mil. | 1956 | 1 |
| Iran | ENTC MNSR | Steady state | Civ. | 1994 | 0.03 |
| Israel | IRR-1 | Steady state | Civ. | 1960 | 5 |
| Italy | RSV Tapiro | Steady state | Civ. | 1971 | 0.005 |
| Jamaica | UWI CNS SLOWPOKE | Steady state | Civ. | 1984 | 0.02 |
| Japan | FCA Tokai research establishm. | Critical assembly | Civ. | 1967 | 0.00002 |
| Japan | KUCA Kyoto University | Critical assembly | Civ. | 1974 | 0.000001 |
| Japan | KUR | Steady state | Civ. | 1964 | 5 |
| Japan | UTR KINKI | Steady state | Civ. | 1961 | 0 |
| Japan | YAYOI | Steady state | Civ. | 1971 | 0.002 |
| Kazakhstan | WWR-K CA | Critical assembly | Civ. | 1972 | 0.0001 |
| Kazakhstan | EWG 1 | Steady state | Civ. | 1972 | 60 |
| Kazakhstan | IGR | Steady state | Civ. | 1961 | 10 |
| Kazakhstan | WWR-K Alatau | Steady state | Civ. | 1967 | 6 |
| Mexico | TRIGA MARK III | Steady state | Civ. | 1968 | 1 |
| Netherlands | LFR | Steady state | Civ. | 1960 | 0.03 |
| Nigeria | NIRR-0001 | Steady state | Civ. | 2004 | 0.03 |
| Pakistan | PARR-2 | Steady state | Civ. | 1989 | 0.03 |
| Poland | MARIA | Steady state | Civ. | 1974 | 30 |
| Russia | BOR-60 | Breeder | Civ. | 1969 | 60 |
| Russia | 659 | Critical assembly | Mil. | 1963 | 0.0001 |
| Russia | 1125 | Critical assembly | Mil. | 1975 | 0.0006 |
| Russia | FM | Critical assembly | Civ. | 1971 | 0.00001 |
| Russia | Astra | Critical assembly | Civ. | 1981 | 0.0001 |
| Russia | BFS-1 | Critical assembly | Civ. | 1961 | |
| Russia | BFS-2 | Critical assembly | Civ. | 1969 | |
| Russia | BR-1 | Critical assembly | Civ. | 1965 | 0.00005 |
| Russia | CA MIR.M1 | Critical assembly | Civ. | 1966 | 0.0001 |
| Russia | Delta | Critical assembly | Civ. | 1985 | 0.0001 |
| Russia | Emphir-2M / EFIR-2M | Critical assembly | Civ. | 1973 | 0.0001 |
| Russia | FM MR | Critical assembly | Civ. | 1971 | 0.0001 |
| Russia | FS-1M | Critical assembly | Civ. | 1970 | 0.000002 |
| Russia | ISKRA | Critical assembly | Civ. | 1996 | 0.0002 |
| Russia | Kvant | Critical assembly | Civ. | 1990 | 0.001 |
| Russia | MAKET | Critical assembly | Civ. | 1976 | |
| Russia | Nartsiss-M2 | Critical assembly | Civ. | 1983 | 0.00001 |
| Russia | FS-2 | Critical assembly | Mil. | 1972 | |
| Russia | FS-4 | Critical assembly | Mil. | 1976 | |

| | | | | | |
|---|---|---|---|---|---|
| Russia | FS-5 | Critical assembly | Mil. | 1987 | |
| Russia | RF-GS | Critical assembly | Civ. | 1962 | 0.00001 |
| Russia | SF-1 | Critical assembly | Civ. | 1972 | 0.0001 |
| Russia | SF-7 | Critical assembly | Civ. | 1975 | 0.0001 |
| Russia | SGO | Critical assembly | Civ. | 1968 | 0.00001 |
| Russia | SO-2M | Critical assembly | Mil. | 1976 | |
| Russia | FM PIK | Critical assembly | Civ. | 1983 | 0.0001 |
| Russia | STEND-2 | Critical assembly | Mil. | 1969 | 0.002 |
| Russia | STEND-3 | Critical assembly | Mil. | 1967 | 0.002 |
| Russia | STEND-4 | Critical assembly | Mil. | 1967 | 0.0005 |
| Russia | STEND-5 | Critical assembly | Mil. | 1967 | 0.0005 |
| Russia | Strela | Critical assembly | Civ. | 1968 | 0.00001 |
| Russia | T2 | Critical assembly | Civ. | 1965 | 2E-07 |
| Russia | BARS-4 | Pulsed | Mil. | 1984 | |
| Russia | BARS-5 | Pulsed | Civ. | 1986 | 0.01 |
| Russia | BARS-6 | Pulsed | Civ. | | |
| Russia | BIGR | Pulsed | Mil. | 1977 | |
| Russia | BIR-2M | Pulsed | Mil. | 1965 | |
| Russia | FBR_L | Pulsed | Civ. | 1981 | |
| Russia | GIR 2 | Pulsed | Mil. | 1993 | |
| Russia | HYDRA (IIN-3M Gidra) | Pulsed | Civ. | 1972 | 0.01 |
| Russia | IBR-2 | Pulsed | Civ. | 1977 | 2 |
| Russia | IBR-30 | Pulsed | Civ. | 1970 | 0.01 |
| Russia | Igrik | Pulsed | Civ. | 1975 | 0.03 |
| Russia | VIR-2M | Pulsed | Mil. | 1980 | |
| Russia | Yaguar (NHUAR) | Pulsed | Civ. | 1990 | 0.01 |
| Russia | ARGUS | Steady state | Civ. | 1981 | 0.02 |
| Russia | BR-10 | Steady state | Civ. | 1958 | 8 |
| Russia | IR-8 | Steady state | Civ. | 1981 | 8 |
| Russia | IRT, MEPhI | Steady state | Civ. | 1967 | 2.5 |
| Russia | IRT-T, Tomsk | Steady state | Civ. | 1967 | 6 |
| Russia | IVV-2M | Steady state | Civ. | 1966 | 15 |
| Russia | MIR.M1 | Steady state | Civ. | 1966 | 100 |
| Russia | OR | Steady state | Civ. | 1989 | 0.3 |
| Russia | RBT-10/2, Minatom | Steady state | Civ. | 1984 | 7 |
| Russia | RBT-6, Minatom | Steady state | Civ. | 1975 | 6 |
| Russia | SM | Steady state | Civ. | 1961 | 100 |
| Russia | WWR-M | Steady state | Civ. | 1959 | 18 |
| Russia | WWR-TS | Steady state | Civ. | 1964 | 15 |
| Russia | R-1 | Critical assembly | Mil. | 1992 | |
| Switzerland | AGN 211 P | Steady state | Civ. | 1959 | 0.002 |
| Syrian Arab Republic | SRR-1 | Steady state | Civ. | 1996 | 0.03 |
| UK | VIPER | Pulsed | Mil. | 1967 | 0.0005 |
| UK | IMPERIAL COLLEGE (CONSORT) | Steady state | Civ. | 1965 | 0.1 |
| Ukraine | WWR-M, Kiev | Steady state | Civ. | 1960 | 10 |
| US | ATRC | Critical assembly | Civ. | 1964 | 0.005 |
| US | Fast-burst FBR | Pulsed | Mil. | 1964 | 10 |
| US | SPR-II | Pulsed | Mil. | 1967 | 0.005 |
| US | SPR-III | Pulsed | Mil. | 1975 | 0.01 |
| US | ACRR | Steady state | Civ. | 1967 | 4 |
| US | ATR | Steady state | Civ. | 1967 | 250 |
| US | HFIR | Steady state | Civ. | 1965 | 85 |
| US | MITR-II | Steady state | Civ. | 1958 | 4.9 |
| US | MURR UNIV. | Steady state | Civ. | 1966 | 10 |
| US | NBSR | Steady state | Civ. | 1967 | 20 |
| US | NRAD | Steady state | Civ. | 1977 | 0.25 |
| US | NTR General Electric | Steady state | Civ. | 1956 | 0.1 |
| US | OSTR, STATE UNIV. | Steady state | Civ. | 1967 | 1.1 |
| US | UWNR UNIV. | Steady state | Civ. | 1961 | 1 |
| US | WSUR, ST.UNIV. | Steady state | Civ. | 1961 | 1 |
| Uzbekistan | Photon | Pulsed | Civ. | 1975 | 1.5 |
| Uzbekistan | WWR-CM, Tashkent | Steady state | Civ. | 1959 | 10 |
| Vietnam | Dalat Research Reactor (DRR) | Steady state | Civ. | 1963 | 0.5 |

The list of facilities is an extract from the HEU Facility Database (HEU-FD) at the Norwegian Radiation Protection Authority (NRPA) covering facilities established after 1978. All illustrations are based on data in HEU-FD. The main source of this database is different versions – 1989 to 2001 – of the IAEA Research Reactor Database and other open-source information as indicated in this paper. For further information on specific information on any of the facilities in Appendix 1 or other relevant facilities, please contact the authors.

# Session 7

# IS Concepts

# A Critical Evaluation of the Critical Time Concept in Nuclear Material Safeguards

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***Abstract:***

*The critical time concept in nuclear material safeguards says that the diversion or illegal use of fissile material by a state has to be detected within a given time, varying between seven to ten days for plutonium or high enriched uranium to three months for spent reactor fuels. Given that such a diversion or illegal use is detected by an inspector within that time, however, it may take months before actions are taken and sanctions are eventually imposed to the state under consideration. How does this fit together?*
*In this paper this problem is analysed with the help of a normal form game with vector-valued payoffs both to the operator and inspector, where the components of the payoff vector take into account both aspects. For a special application a simple solution is given and discussed.*

**Keywords:** Critical time; detection time; unannounced interim inspections; inspection strategies

## 1. Introduction

In the course of the discussions during the IAEA Safeguards Symposium in Vienna in September 2006, the following problem was raised: The critical time concept in nuclear material safeguards says that the diversion or illegal use of fissile material by a state has to be detected within a given time, varying between seven to ten days for plutonium or high enriched uranium to three months for spent reactor fuels, see IAEA [1]. Given that such a diversion or illegal use is detected by an inspector within that time, however, it may take months before actions are taken and sanctions are eventually imposed to the state under consideration. How does this fit together?
In the spirit of the critical time concept, the inspection problem has to be modelled such that the inspector has won or lost if he has detected the illegal action in time, whereas just the opposite holds for the state under consideration. In the spirit of the actions to be taken by the inspectorate on the other hand, any illegal action should simply be detected as early as possible. Thus it is quite natural to model the inspection problem in such a way, that these two different objectives are taken into account simultaneously: We are let to a non-cooperative two-person game with vector-valued payoffs.
This generalized concept is illustrated with the help of an unannounced interim inspection problem: A spent fuel storage is considered where once a year a physical inventory (PIV) is taken, and at most three unannounced interim inspections are performed within one year. We assume that both players behave non-sequentially, i.e., fix their decisions at the beginning of the year, therefore, in the next section we model our problem by a normal form game with vector valued payoffs. A short outline of the analysis if such games is given in the third section since it is not contained in introductory texts and at least according to our knowledge, has not yet been applied in safeguards. For the case of just one interim inspection a complete solution of the problem, i.e., equilibrium strategies and payoffs, is presented in the third section. The application of these results to the practice of inspections is discussed in the concluding section; it turns out to be surprisingly simple, at least for the case under consideration.

## 2. Problem Formulation

Let us start by formulating separate games for the two aspects of the problem raised in the introduction, which we call the “critical time game” and the “playing for time game”. Only after having introduced these two games we are able to formulate the game which addresses our original problem.

### 2.1. The critical time game

The critical time game for an interim storage with one intermediate inspection may be described as follows: The inspector can perform his intermediate inspection after multiples of the critical time, i.e., after three, six or nine month after the last physical inventory. In the following we call these time points *1*,*2* and *3*. After 12 month (time point *4*) again a physical inventory is taken. For the operator – in case he will act illegally – it also makes no sense to start an illegal action somewhere, but only at the beginning of the reference time or immediately after the intermediate inspection. If the inspection takes places within the critical time after the start of the illegal action, the inspector has won and the operator has lost, otherwise vice versa. The payoff to the two players (operator, inspector) are

$(d_1, -c_1)$ for detection of an illegal action outside the critical time, and
$(-b_1, -a_1)$ for detection of an illegal action within the critical time.

Here we assume $b_1 > 0, d_1 > 0$ and $0 < a_1 < c_1$, since we normalize the payoffs for legal behaviour to zero, and timely detection of an illegal action still being worse then legal behaviour of the operator. The normal form of this “illegal” game, i.e., the game where the legal behaviour of the operator is excluded, is represented in Table 1, where the rows resp. the columns represent the pure strategies of the operator resp. the inspector, and where in the lower left resp. upper right corner of each entry the payoffs to the operator resp. the inspector are given. For an introduction to normal form games, the Nash equilibrium concept and solution methods see, e.g., Myerson [2] and Nash [3]. Moreover, in section 3 a short introduction into the more advanced formalism needed there is given.

| Op \ Insp | *1* | *2* | *3* |
|---|---|---|---|
| *0* | $-a_1$ / $-b_1$ | $-c_1$ / $d_1$ | $-c_1$ / $d_1$ |
| *1* | $-c_1$ / $d_1$ | $-a_1$ / $-b_1$ | $-c_1$ / $d_1$ |
| *2* | $-c_1$ / $d_1$ | $-c_1$ / $d_1$ | $-a_1$ / $-b_1$ |
| *3* | $-a_1$ / $-b_1$ | $-a_1$ / $-b_1$ | $-a_1$ / $-b_1$ |

**Table 1:** Normal form of the critical time game

We realize immediately that the fourth pure strategy of the operator is dominated thus, we have to consider a quadratic $3 \times 3$ game. The Nash equilibrium of this game consists in mixed strategies, i.e., probabilities with which the pure strategies are played. Using the symmetry of the $3 \times 3$ matrix we obtain for the operator and for the inspector

$$q_0^* = q_1^* = q_2^* = \frac{1}{3}, q_3^* = 0 \qquad \text{and} \qquad p_1^* = p_2^* = p_3^* = \frac{1}{3},$$

independent of the payoff parameters; with the expected payoffs to the operator and inspector

$$Op^* = \frac{1}{3}(2\,d_1 - b_1) \qquad \text{and} \qquad Insp^* = -\frac{1}{3}(2\,c_1 + a_1) \quad .$$

So far we discussed only the illegal game, i.e., the game in which the operator acts illegally with certainty. He will behave legally, if his expected payoff in this case is larger than in the other. Thus, if we add to the game in Table 1 the legal behaviour of the operator as his fifth pure strategy, and since the operator´s expected payoff in case of legal behaviour is zero, for $2\,d_1 - b_1 < 0$ the equilibrium strategy of the operator is legal behaviour, i.e.,

$$q_0^* = q_1^* = q_2^* = q_3^* = 0\,,\; q_4^* = 1\,.$$

The corresponding equilibrium strategy of the inspector is not unique, as the Nash conditions for the operator show:

$$0 \geq -b_1\; p_1^* + d_1\; p_2^* + d_1\; p_3^*$$
$$0 \geq d_1\; p_1^* - b_1\; p_2^* + d_1\; p_3^*$$
$$0 \geq d_1\; p_1^* + d_1\; p_2^* - b_1\; p_3^*$$

or equivalently

$$p_j^* \geq \frac{1}{1 + b_1/d_1},\; j = 1,2,3\,, \quad p_1^* + p_2^* + p_3^* = 1\,.$$

Because of $2\,d_1 - b_1 < 0$ we have $1 + b_1/d_1 > 3$, which means that the equilibrium strategy for the illegal game is contained in the set of equilibrium strategies of the legal game.

### 2.2. The playing for time game

Consider now the playing for time game. The pure strategies of both players are the same as before, whereas the outcome for the operator is the better, the longer the time between the start of the illegal action and the detection, and for the inspector the better the shorter. Since an illegal action will be detected anyhow at the end of the reference time, he gets a penalty $-b_2$ thus, the payoff to the two players (operator, inspector) are now

$$(\,d_2 \cdot i - b_2, -c_2 \cdot i - a_2\,)$$

for the time $i$ $(i = 1, 2, 3)$ between the start of the illegal action and its detection. Here we assume $d_2 > 0, b_2 > 0$ and $c_2 > 0, a_2 > 0$. The normal form of this game is given in Table 2; we see that the fourth pure strategy of the operator is again dominated.

| Op \ Insp | *1* | *2* | *3* |
|---|---|---|---|
| *0* | $-c_2 - a_2$<br>$d_2 - b_2$ | $-2\,c_2 - a_2$<br>$2\,d_2 - b_2$ | $-3\,c_2 - a_2$<br>$3\,d_2 - b_2$ |
| *1* | $-3\,c_2 - a_2$<br>$3\,d_2 - b_2$ | $-c_2 - a_2$<br>$d_2 - b_2$ | $-2\,c_2 - a_2$<br>$2\,d_2 - b_2$ |
| *2* | $-2\,c_2 - a_2$<br>$2\,d_2 - b_2$ | $-2\,c_2 - a_2$<br>$2\,d_2 - b_2$ | $-c_2 - a_2$<br>$d_2 - b_2$ |
| *3* | $-c_2 - a_2$<br>$d_2 - b_2$ | $-c_2 - a_2$<br>$d_2 - b_2$ | $-c_2 - a_2$<br>$d_2 - b_2$ |

**Table 2:** Normal form of the playing for time game

Again there is a Nash equilibrium in mixed strategies. With the help of a *Mathematica*®-program developed by Canty [4] we obtain for the operator and for the inspector

$$q_0^* = \frac{1}{3}, \quad q_1^* = \frac{1}{6}, \quad q_2^* = \frac{1}{2}, \quad q_3^* = 0 \qquad \text{and} \qquad p_1^* = \frac{1}{3}, \quad p_2^* = \frac{1}{2}, \quad p_3^* = \frac{1}{6},$$

again independent of the payoff parameters. The expected equilibrium payoffs are

$$Op^* = \frac{11}{6}\, d_2 - b_2 \qquad \text{and} \qquad Insp^* = -\frac{11}{6}\, c_2 - a_2 \quad .$$

With the same arguments as before we note that for $11\, d_2 - 6\, b_2 < 0$ the equilibrium strategy of the operator is legal behaviour; the corresponding equilibrium strategy of the inspector has the same properties as that for the critical time game.

## 2.3. The vector-valued game

Let us now turn to our original problem, namely to consider simultaneously the critical time game and the playing for time game. Since in both games the pure strategies of both players are the same, we just have to consider the payoffs simultaneously which leads to a vector-valued game the normal form of which is given in Table 3.

| Op \ Insp | 1 | 2 | 3 |
|---|---|---|---|
| 0 | $\begin{pmatrix} -a_1 \\ -c_2 - a_2 \end{pmatrix}$ $\begin{pmatrix} -b_1 \\ d_2 - b_2 \end{pmatrix}$ | $\begin{pmatrix} -c_1 \\ -2\,c_2 - a_2 \end{pmatrix}$ $\begin{pmatrix} d_1 \\ 2\,d_2 - b_2 \end{pmatrix}$ | $\begin{pmatrix} -c_1 \\ -3\,c_2 - a_2 \end{pmatrix}$ $\begin{pmatrix} d_1 \\ 3\,d_2 - b_2 \end{pmatrix}$ |
| 1 | $\begin{pmatrix} -c_1 \\ -3\,c_2 - a_2 \end{pmatrix}$ $\begin{pmatrix} d_1 \\ 3\,d_2 - b_2 \end{pmatrix}$ | $\begin{pmatrix} -a_1 \\ -c_2 - a_2 \end{pmatrix}$ $\begin{pmatrix} -b_1 \\ d_2 - b_2 \end{pmatrix}$ | $\begin{pmatrix} -c_1 \\ -2\,c_2 - a_2 \end{pmatrix}$ $\begin{pmatrix} d_1 \\ 2\,d_2 - b_2 \end{pmatrix}$ |
| 2 | $\begin{pmatrix} -c_1 \\ -2\,c_2 - a_2 \end{pmatrix}$ $\begin{pmatrix} d_1 \\ 2\,d_2 - b_2 \end{pmatrix}$ | $\begin{pmatrix} -c_1 \\ -2\,c_2 - a_2 \end{pmatrix}$ $\begin{pmatrix} d_1 \\ 2\,d_2 - b_2 \end{pmatrix}$ | $\begin{pmatrix} -a_1 \\ -c_2 - a_2 \end{pmatrix}$ $\begin{pmatrix} -b_1 \\ d_2 - b_2 \end{pmatrix}$ |
| 3 | $\begin{pmatrix} -a_1 \\ -c_2 - a_2 \end{pmatrix}$ $\begin{pmatrix} -b_1 \\ d_2 - b_2 \end{pmatrix}$ | $\begin{pmatrix} -a_1 \\ -c_2 - a_2 \end{pmatrix}$ $\begin{pmatrix} -b_1 \\ d_2 - b_2 \end{pmatrix}$ | $\begin{pmatrix} -a_1 \\ -c_2 - a_2 \end{pmatrix}$ $\begin{pmatrix} -b_1 \\ d_2 - b_2 \end{pmatrix}$ |

**Table 3:** Vector valued normal form game of the original problem

We can see that the components of the operator´s payoff if he plays his 3rd pure strategy are always greater or at least equal to the corresponding components if he plays his fourth pure strategy, i.e., the fourth pure strategy of the operator is again dominated. The analysis of this game still poses problems, since the payoffs of the two separate games cannot be compared directly. For example is $d_1$ an absolute gain of the operator, whereas $d_2$ is the rate of a gain. Therefore, we make the additional assumptions that the maximal and minimal payoffs to both players are the same. We justify this with the remark that both games describe the same principal conflict situation thus, the best and the worst outcome is evaluated in the same way by the two players. This leads to the following conditions

$$d_2 - b_2 = -b_1\,, \quad 3\,d_2 - b_2 = d_1\,, \quad -c_2 - a_2 = -a_1\,, \quad -3\,c_2 - a_2 = -c_1$$

which is equivalent to

$$d_2 = \frac{1}{2}(b_1 + d_1), \quad b_2 = \frac{1}{2}(3b_1 + d_1), \quad c_2 = \frac{1}{2}(c_1 - a_1), \quad a_2 = \frac{1}{2}(3a_1 - c_1).$$

Because of $b_1 > 0$, $d_1 > 0$ and $a_1 > 0$, $c_1 > 0$ the following conditions have to be fulfilled

$$0 < d_2 < b_2 < 3d_2 \qquad \text{and} \qquad 0 < a_2 < c_2 < 3a_2 .$$

Taking these conditions into account we can model the conflict situation as the vector-valued normal form game given in Table 4, where we have already deleted the fourth pure strategy of the operator, as argued above.

| Op \ Insp | 1 | 2 | 3 |
|---|---|---|---|
| 0 | $\begin{pmatrix}-a_1\\-a_1\end{pmatrix}$ $\begin{pmatrix}-b_1\\-b_1\end{pmatrix}$ | $\begin{pmatrix}-c_1\\-\frac{1}{2}(c_1+a_1)\end{pmatrix}$ $\begin{pmatrix}d_1\\\frac{1}{2}(d_1-b_1)\end{pmatrix}$ | $\begin{pmatrix}-c_1\\-c_1\end{pmatrix}$ $\begin{pmatrix}d_1\\d_1\end{pmatrix}$ |
| 1 | $\begin{pmatrix}-c_1\\-c_1\end{pmatrix}$ $\begin{pmatrix}d_1\\d_1\end{pmatrix}$ | $\begin{pmatrix}-a_1\\-a_1\end{pmatrix}$ $\begin{pmatrix}-b_1\\-b_1\end{pmatrix}$ | $\begin{pmatrix}-c_1\\-\frac{1}{2}(c_1+a_1)\end{pmatrix}$ $\begin{pmatrix}d_1\\\frac{1}{2}(d_1-b_1)\end{pmatrix}$ |
| 2 | $\begin{pmatrix}-c_1\\-\frac{1}{2}(c_1+a_1)\end{pmatrix}$ $\begin{pmatrix}d_1\\\frac{1}{2}(d_1-b_1)\end{pmatrix}$ | $\begin{pmatrix}-c_1\\-\frac{1}{2}(c_1+a_1)\end{pmatrix}$ $\begin{pmatrix}d_1\\\frac{1}{2}(d_1-b_1)\end{pmatrix}$ | $\begin{pmatrix}-a_1\\-a_1\end{pmatrix}$ $\begin{pmatrix}-b_1\\-b_1\end{pmatrix}$ |

**Table 4:** Modified vector valued normal form game

## 3. Vector valued games in normal form

In order to analyse the vector-valued normal form game formulated in section 2.3, we have to introduce the concept of the Pareto equilibrium, which is a natural generalization of the Nash equilibrium concept.

For vectors $Z = (z_1, z_2, \ldots, z_n)$ and $W = (w_1, w_2, \ldots, w_n)$ we define the natural semi-order in $R^n$ by $W \succ Z$, i.e., $W$ dominates $Z$, if and only if $w_i \geq z_i$ for all $i$ $(i = 1, \ldots, n)$ and $w_i > z_i$ for at least one $i$ $(i \in \{1, \ldots, n\})$, see, e.g., Yu [5]. For a set $A \subseteq R^n$ we define

$$M(A) := \{ Z \in A : \text{ no } W \in A \text{ exists with } W \succ Z \}$$

as the set of the *undominated* elements of $A$. It can be shown that $Z \in M(A)$ if and only if

$$A \cap \{ W \in R^n : w_i \geq z_i, i = 1, \ldots, n \} = Z ,$$

which represents a geometrical interpretation of $M(A)$, see, e.g., Borm et. al. [6].

The vector-valued normal form game considered in this paper is a game with two players, namely the inspector and the operator, and each of the players has a finite set of pure strategies. The operator´s set of pure strategies consists of all time points, where he can start his illegal action, i.e., $\Phi_1 := \{ 0, 1, 2, 3 \}$. The inspectors´s set of pure strategies consists of all time points, where he can

perform his inspection, i.e., $\Phi_2 := \{1, 2, 3\}$. Given a pure strategy combination $(i, j) \in \Phi_1 \times \Phi_2$, the operator will receive the payoff vector $Op(i, j) \in R^2$ and the inspector the payoff vector $Insp(i, j) \in R^2$. The games in section 2.1 and 2.2 are special cases of vector-valued normal form games; the players receive a real number as payoffs instead of a vector. In order to handle those kind of games described in the previous section, we also have to introduce the concept of mixed strategies. A mixed strategy of a player is a probability distribution over his set of pure strategies. For our games that means, that the operator´s mixed strategies have the structure

$$q := (q_0, q_1, q_2, q_3) \quad \text{with} \quad q_i \geq 0, i = 0, 1, 2, 3 \quad \text{and} \quad \sum_{i=0}^{3} q_i = 1,$$

where $q_i$ is the probability, that the operator starts his illegal action at time point $i$. For the inspector we get

$$p := (p_1, p_2, p_3) \quad \text{with} \quad p_j \geq 0, j = 1, 2, 3 \quad \text{and} \quad \sum_{j=1}^{3} p_j = 1,$$

where $p_j$ is the probability, that the inspector performs his inspection at time point $j$.

Having introduced the concept of mixed strategies, the vector-valued payoff function of each player, originally defined on the set of pure strategy combinations $\Phi_1 \times \Phi_2$, can be extended to the set of mixed strategies in the following way:

$$Op(q, p) := \sum_{i=0}^{3} \sum_{j=1}^{3} q_i \; p_j \; Op(i, j) \quad \text{and} \quad Insp(q, p) := \sum_{i=0}^{3} \sum_{j=1}^{3} q_i \; p_j \; Insp(i, j).$$

$Op(q, p)$ respectively $Insp(q, p)$ is called the expected payoff vector to the operator respectively to the inspector.

Let us now introduce the Pareto equilibrium concept. We consider the mixed extension of a vector-valued normal form game. Then $(q^*, p^*)$ is a Pareto equilibrium of the game if and only if

$$Op(q^*, p^*) \in M(Op(q, p^*)) \quad \text{and} \quad Insp(q^*, p^*) \in M(Insp(q^*, p)).$$

This definition says, that any unilateral deviation of one player from the mixed strategy combination $(q^*, p^*)$ will not improve his expected payoff vector. In case of real-valued payoff´s to each player (see the games in Tables 1 and 2), the set of Pareto equilibria is equal to the set of Nash equilibria however, there are important differences between real- and vector-valued normal form games which are discussed by Krieger [7,8].

The Pareto equilibrium was first introduced by Shapley [9] – he called it strong equilibrium - for two-person vector-valued zero-sum normal form games. Borm et. al. [6] generalized Shapley´s idea considering strong equilibria for two-person vector-valued normal form games. A further generalization to strong equilibria for *n*-person vector-valued normal form games was made by Krieger [7,8].

It is quite difficult to determine the Pareto equilibria of vector-valued normal form games by using the definition of these equilibria. Therefore, we present a Lemma which shows a very close relation between Pareto equilibria in vector-valued normal form games and Nash equilibria in certain induced real-valued normal form games.

Let us consider the mixed extension of a (in our case: two-component) vector-valued normal form game, and define

$$\Delta := \left\{ a = (a_1, a_2)^T \in R^2 : a_1 \geq 0, a_2 \geq 0 \quad \text{and} \quad a_1 + a_2 = 1 \right\}.$$

Then the following Lemma holds:

1. Consider arbitrary vectors $\lambda, \gamma \in \Delta$ : If the mixed strategy combination $(q^*, p^*)$ is a Nash equilibrium of the (real-valued) normal form game with expected payoffs $\lambda^T Op(q,p)$ for the operator and $\gamma^T Insp(q,p)$ for the inspector, then $(q^*, p^*)$ is a Pareto equilibrium of the original game.
2. If the mixed strategy combination $(q^*, p^*)$ is a Pareto equilibrium of the original game, then there exist vectors $\lambda, \gamma \in \Delta$ such that $(q^*, p^*)$ is a Nash equilibrium of the (real-valued) normal form game with expected payoffs $\lambda^T Op(q,p)$ for the operator and $\gamma^T Insp(q,p)$ for the inspector.

The proof of this Lemma may be found in Borm et. al. [6] or a generalization in Krieger [8]. Using this Lemma and Nash´s famous existence theorem, see [3], we get the result, that every vector-valued normal form game has at least one Pareto equilibrium in mixed strategies.

## 4. Solution of the vector-valued game

Using the payoff matrix in Table 4 the operator´s payoff vectors are multiplied with the scalarization vector $(\gamma, 1-\gamma)^T$ $(\gamma \in [0,1])$ and the inspectors payoff vectors with the scalarization vector $(\lambda, 1-\lambda)^T$ $(\lambda \in [0,1])$. This leads with the abbreviations

$$a := \frac{1}{2}[(1+\gamma)\, d_1 - (1-\gamma)\, b_1] \qquad \text{and} \qquad b := -\frac{1}{2}[(1+\lambda)\, c_1 + (1-\lambda)\, a_1]$$

to the real-valued normal form game given in Table 5:

| Op \ Insp | 1 | 2 | 3 |
|---|---|---|---|
| 0 | $-a_1$<br>$-b_1$ | $b$<br>$a$ | $-c_1$<br>$d_1$ |
| 1 | $-c_1$<br>$d_1$ | $-a_1$<br>$-b_1$ | $b$<br>$a$ |
| 2 | $b$<br>$a$ | $b$<br>$a$ | $-a_1$<br>$-b_1$ |

**Table 5:** Scalarized form of the vector-valued normal form game given in Table 4

Because of the many – in total six – parameters of this game it is not so easy to use Canty´s program, see [4], for the determination of its Nash equilibria. It is better instead, to use the method of making the adversary indifferent as regards to his strategies to be chosen. This way we obtain for fixed values of $\gamma$ and $\lambda$ the only Nash equilibrium in mixed strategies of this game as follows. The equilibrium strategies of the operator and the inspector are

$$q_0^*(\lambda) = \frac{1+\lambda}{(1+\lambda)^2+2}, \quad q_1^*(\lambda) = \frac{1}{2}\frac{(1+\lambda)^2}{(1+\lambda)^2+2}, \quad q_2^*(\lambda) = \frac{1}{2}\frac{3+\lambda^2}{(1+\lambda)^2+2}$$

$$p_1^*(\gamma) = \frac{1+\gamma}{(1+\gamma)^2+2},\quad p_2^*(\gamma) = \frac{1}{2}\frac{3+\gamma^2}{(1+\gamma)^2+2},\quad p_3^*(\gamma) = \frac{1}{2}\frac{(1+\gamma)^2}{(1+\gamma)^2+2}$$

and the expected equilibrium payoffs (of the scalarized game)

$$Op^*(\gamma) = (d_1+b_1)\left(f(\gamma) - \frac{1}{1+d_1/b_1}\right) \quad \text{and} \quad Insp^*(\lambda) = -(c_1-a_1)\left(f(\lambda) - \frac{1}{1-c_1/a_1}\right)$$

where $f(x)$ defined on [0,1] is given by

$$f(x) := \frac{1+x}{2} - \frac{1}{4}\,\frac{(1+x)^3}{(1+x)^2+2}\ .$$

According to the Lemma in the second section, the complete set of Pareto equilibria is given by the set of these Nash equilibria for all combinations $(\lambda,\gamma) \in [0,1]\times[0,1]$.

As we see the equilibrium strategies of both players depend only on $\gamma$ respectively $\lambda$; more than that, $q_0^*(\lambda)$ and $p_1^*(\gamma)$ depend on $\lambda$ respectively $\gamma$ very slightly. For $p_1^*(\gamma)$ for example we have

$$p_1^*(0) = p_1^*(1) = \frac{1}{3}$$

and the only maximum of $p_1^*(\gamma)$ in [0,1] is given at

$$\gamma_0 = \sqrt{2} - 1 \approx 0.412 \qquad \text{and} \qquad p_1^*(\gamma_0) = \frac{\sqrt{2}}{4} \approx 0.353\,.$$

The function $f(x)$ determining the equilibrium payoffs is monotonely increasing from $f(0) = 5/12$ to $f(1) = 2/3 = 8/12$. Recall that $\lambda = \gamma = 0$ corresponds to the playing for time game and $\lambda = \gamma = 1$ to the critical time game.

Let us continue as in the previous games: So far we discussed only the illegal game, i.e., the game in which the operator acts illegally with certainty. We assume that the operator will behave legally if his expected payoff vector in case of legal behaviour is larger than in case of illegal behaviour in both components. Since the operator´s payoff vector in case of legal behaviour is zero in both components, the condition for legal behaviour is:

$$Op^*(\gamma) < 0 \qquad \text{or} \qquad f(\gamma) < \frac{1}{1+b_1/d_1} \qquad \text{for all} \qquad \gamma \in [0,1]\,.$$

Since $f(\gamma)$ is a monotonely increasing function, this condition can be assured if we postulate

$$f(1) < \frac{1}{1+b_1/d_1} \qquad \text{or explicitly} \qquad 2\,d_1 < b_1\,.$$

Like in the games considered in sections 2.1 and 2.2 the equilibrium strategy of the inspector is not unique, as the Nash conditions for the operator show (for fixed $\gamma$):

$$0 \geq -b_1\ p_1^* + a\ p_2^* + d_1\ p_3^*$$

$$0 \geq d_1\ p_1^* - b_1\ p_2^* + a\ p_3^*$$
$$0 \geq a\ (\ p_1^* + p_2^*\ ) - b_1\ p_3^*\ ,$$

where $a$ has been defined before.

We see that these conditions for the operator are fulfilled for any $\gamma \in [0,1]$ if they are fulfilled for $\gamma = 1$. This means that the inspector is on the safe side if he plays the critical time game.

One can understand this simple result if one realizes that, if the payoffs of the critical and the playing for time game are adjusted in the way we did it, the condition for legal behaviour in the critical time game implies that condition in the playing for time game.

## 5. Discussion

What practical conclusions do we draw from these results for the inspector, who has to choose a concrete (mixed) inspection strategy? First, we see that the inspection probabilities vary only slightly with $\gamma$, which means that in no case a large mistake will be made if an inspection strategy with a specific $\gamma \in [0,1]$ is taken.

Second, if we are able to use the criterion of inducing the operator to legal behaviour, then the answer to our basic problem is clear: For $2\ d_1 < b_1$ the operator will behave legally, and the appropriate equilibrium strategy of the inspector is his equilibrium strategy for the illegal equilibrium and $\gamma = 1$, since this way, as pointed out, he is on the safe side.

If the condition $2\ d_1 < b_1$ is not fulfilled, or if quite generally the payoff parameters $d_1$ and $b_1$ are not so well known, a so called equilibrium selection problem arises if one wants to recommend a concrete inspection strategy. We do not want to enter here into this difficult area of modern game theory, but use instead a pragmatic, so to speak decision theoretical approach: The inspector takes a pessimistic point of view and chooses that inspection strategy that maximizes the operator´s payoff. This way again we are led to the conclusion just to consider the critical time game.

We asked in the introduction “How does this fit together?”. While our answer is simple and clear in a technical sense, i.e., as regards to what the inspector should do, its interpretation is not so easy. Therefore, we just state that according to our analysis only the critical time concept matters, and that the administrative delays of actions and sanctions to be taken after detected illegal action are regrettable and should be minimized, but do not change the original concept.

Let us conclude with two remarks: Of course our result depends strongly on the assumptions, i.e., on the concrete inspection problem considered here. No guesses can be made as to the results for other cases – let us only remember that our choice of consistent payoff parameters depends on the critical time lengths and the frequency of inspections.

Finally, our analysis demonstrates in which way problems with multiple objectives (payoffs) are solved, which are ubiquitous also in safeguards and, as we mentioned in the introduction, according to our best knowledge have not yet been applied here.

## 6. Acknowledgement

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# Safeguards Mailboxes and Unannounced Inspections

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***Abstract:***

*Under integrated safeguards, new State-level approaches will be implemented. One important tool is the use of a safeguards mailbox in conjunction with unannounced inspections. Under traditional safeguards, the transfer of used fuel to dry storage requires a large and growing IAEA inspector effort. Under integrated safeguards, transfers to dry storage are most likely to benefit from the mailbox/unannounced inspection approach.*

*The arrangement is simple in concept. The operator is required to send advance information to the IAEA on the planned activities and then proceeds according to the schedule. The IAEA has the option to send an inspector unannounced to observe the activity.*

*In practice, setting up the required procedures is more complicated. The means of communicating the operator's intentions has to be sent in a secure and trusted way. Also on-site procedures are required so that the inspector may reach the location of the activity within an agreed time window and have the agreed operator support.*

*This paper reviews the Canadian experiences with the safeguards mailbox and unannounced inspections with particular reference to transfers to dry storage. The electronic mail box design developed for this purpose will be described and some information on the possible reductions on inspector effort will be provided.*

**Keywords:** *integrated safeguards; mailbox; unannounced inspection.*

## 1. Introduction

Guided by a State-Level Approach, new methods are being used. One important tool is the use of a safeguards mailbox in conjunction with unannounced inspections. The arrangement is simple in concept. The operator is required to send advance information to the IAEA on the planned activities and then proceeds with those planned activities without waiting for the IAEA inspector. The IAEA then has the option to send an inspector unannounced to verify any of these activities.

A safeguards mailbox can be implemented using various means including regular mail and fax transmission as well as electronic mail. A mailbox may be physically located at the facility, the IAEA or perhaps another location such as a Regional Office. Due to the safeguards implications, special features need to be implemented in the mailbox to assure the sender that the message has been securely received and to assure the IAEA that the message is valid.

These features are not provided by regular electronic mail. However, a system has been devised that allows the submitting party to use the secure mail available on a regular desk-top computer. Custom functionality has been implemented in the IAEA's receiving mailbox so that it can promptly issue a secure confirmation to the sender and ensure that the responsible IAEA safeguards officer is alerted to the new information.

## 2. Transfers to dry storage

The evolution of safeguards and in particular the advent of integrated safeguards has allowed new options to be considered to meet safeguards objectives by using more effective and/or more efficient methods. In particular, a move from "mechanistic" measures to random and unannounced inspections has been proposed. By replacing the traditional announced inspections and implementing random unannounced inspections, the IAEA can have enhanced confidence in their observations and confidence that the advance information and declarations provided by the operator are true. In this respect, the mailbox is an essential partner for unannounced inspections.

In Canada's case, the IAEA effort required to oversee transfers to dry storage under traditional safeguards was large and getting larger. It had been evident for some time that a more efficient method of confirming compliance was needed. A trial of the Unannounced Inspection and Mailbox concept was carried out in 2004 [1].

### 2.1. Traditional safeguards

For spent fuel, there are two types of dry storage in use in Canada:

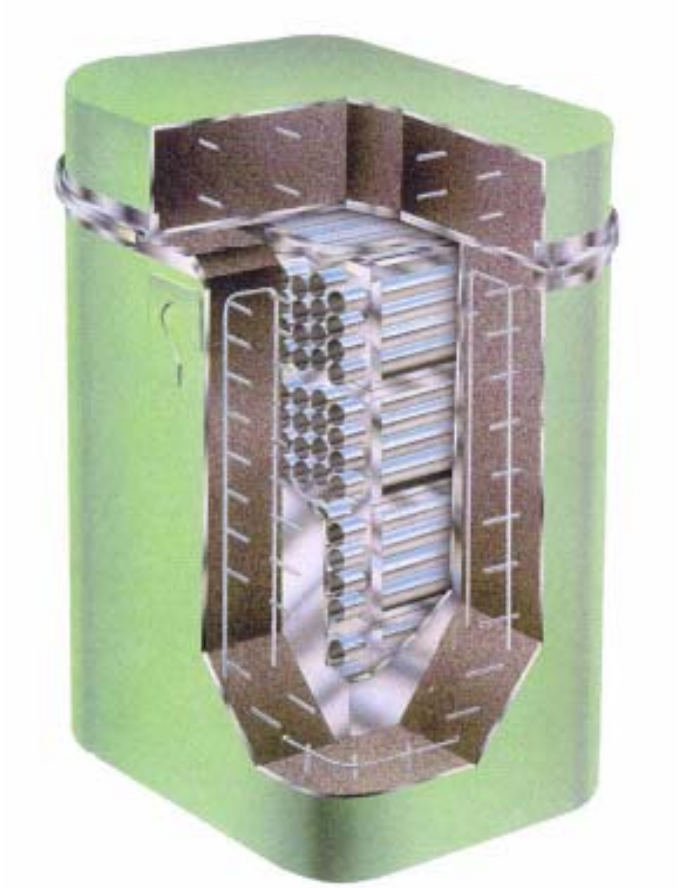
- Dry Storage Containers (DSCs) – large but movable steel and concrete containers (Figure 1)
- Silos – fixed structures with baskets containing the used fuel bundles stacked inside.

The DSCs include the largest amount of spent fuel to be stored in Canada and will be the main focus of the current discussion. This storage system is used by all the multi-unit stations. DSCs are loaded with spent fuel in the spent fuel bay and then transferred to another location (Figure 2) for permanent closure by welding and then to long term storage. Typical transfer rates for the multi-unit stations are one to four DSC's per week. Under traditional safeguards, the removal of spent fuel from the bay and the transfer of the temporarily closed DSC to the Dry Storage Facility required the presence of an IAEA inspector at each major step of the transfer. Not only would this take considerable inspector effort, but scheduling accurately in advance was a challenge and has led to ineffective use of both the inspector's time and that of the facility's personnel.

### 2.2. Integrated safeguards

Under integrated safeguards, facilities will send advance information on planned transfers to dry storage to the IAEA and accept Unannounced Inspections. The advance information would need to be updated when the schedule changes. The IAEA inspector would have the option to arrive unannounced at the facility to verify that the activities conform to the latest information provided by the facility. Special procedures are being set up to ensure the inspector has access to the relevant locations in a timely fashion. Reference [2] provides more detail on the operations involved. Under these arrangements it is expected that PDI (Person Day of Inspection) requirements can be reduced to a fraction of those needed under traditional safeguards.

Unlike under traditional safeguards, the facility will have no information on whether an inspector will visit or at what time; they must expect an inspector to arrive unannounced. Un-notified activities could severely undermine confidence in the operator. Any activities or schedule provided in the advance information will need to be adhered to or sufficiently accounted for. Should circumstances change, updated information must be provided to the Mailbox. For these reasons confidence in the Mailbox system is crucial. Time stamping and the means to authenticate the messages are critical to ensure there are no disputes. This will come particularly important when changes to the schedule have to be made when the inspector is already on his way to the facility to carry out an unannounced inspection.



**Contains:**

4 modules

384 bundles

**Weight full:**

70 tonnes (70 Mg)

**Weight empty:**

60 tonnes (60 Mg)

**Figure 1 – Dry Storage Container (DSC) used by Canadian Multi-unit facilities**

**Figure 2 - A loaded DSC being transported**

## 3. Requirements for safeguards use of a mailbox

The specialised role of the mailbox when used in conjunction with unannounced inspections requires that special attention be paid to how the IAEA mailbox functions. A high level of trust with all parties must be established in the communications. Reference [3] provides a detailed analysis of the mailbox requirements from the perspectives of both the IAEA and the operator.

The IAEA will need to be confident that:

- Any message sent to the mailbox cannot subsequently be denied by the operator (non-repudiation).
- The time that the message was sent is true (trusted time stamp)
- The message cannot be altered following its initial transmission (in-alterability)
- The person sending the message is who he says he is and is authorized to submit information for that facility.

The operator will need:

- Assurance that no unauthorized party can make a submission in his name (counterfeit not possible)
- An acknowledgement to be sure that the information has been received. (trusted acknowledgement)
- A timely acknowledgement from the IAEA (within a minute)
- Assurance that sensitive information is only available to the parties that need to know (confidentiality)
- The ability to provide to the IAEA inspector any recent message sent to the mailbox while the inspector was en-route to the inspection site. The authenticity of this message must be digitally verifiable by the inspector.

The SSAC will require:

- A copy of all of the operator's submissions to the mailbox and the corresponding acknowledgements
- Assurance that the original message and its acknowledgement are genuine.

The general contents of messages sent to the mailbox should be specified in the procedure agreed by the IAEA and the SSAC. The advance information is expected to contain incremental updates to an annual schedule for transfers to dry storage and associated activities. Other submissions may include declarations on the current status of transfer operations and material holdings.

## 4. The IAEA mailbox

It was first thought that these specialised requirements could only be fully met using custom stand-alone systems both for the operator and for the IAEA. But there were concerns that custom systems incorporated at facilities would be difficult to implement and support, particularly so if widely deployed at a whole range of facility types. Who would be responsible if it did not function properly? Another problem with this solution was that it may be difficult to determine which individual made a particular declaration.

A solution has been devised (Figure 3) that depends on the functionality of a standard desk top computer and standard mail systems; it is based on standard S/MIME [5] and associated technical protocols. This solution meets the needs for the bulk of commercial facilities. The IAEA mailbox that receives the advance information needs some additional functionality. Besides generating an acknowledgement automatically, the IAEA mailbox also has the ability to recognize senders and distribute their messages according to their associated MBA and responsible IAEA inspector.

Initial trials of the mailbox concept carried out in May 2004 [1] allowed an early test of electronic mail. One problem that was immediately apparent was the time taken for e-mail to reach an IAEA address (up to twenty minutes). It was thought un-acceptable for the operator to wait this length of time for an acknowledgement. Adjustments had to be made to ensure the IAEA mailbox traffic was delivered promptly to the appropriate mail server and the acknowledgement promptly delivered to the originator. Currently, acknowledgements are delivered within a minute.

## 5. Submissions to the mailbox

The operator needs to be able to send secure mail using his regular desk-top computer and e-mail software. To send mail securely, he must be in possession of a public key certificate; ideally issued through his own organization's Public Key Infrastructure (PKI). The operator's message can be sent as plain text within the e-mail[i] or as an attachment to the e-mail; both the plain text and the attachment will be signed and encrypted. The attachment can be prepared using any standard application such as Word, Excel or Acrobat. The use of standardized formats such as can be provided with Excel or XML opens up the possibility of the IAEA further automating the handling of these messages. Mail systems such as Outlook will automatically handle the authentication and encryption of the text body and any attachments as well as the reverse processes. It should be noted however that the subject line of the e-mail is not encrypted nor included in the signature, so no confidential information should be placed there.

Before sending information to the mailbox for the first time, the sender must:

- Be known to the IAEA mailbox
- Have exchanged certificates with the IAEA Mailbox

The SSAC needs to provide the IAEA with a list of individuals responsible for sending messages and their MBA responsibilities; this list is made available to the Mailbox software and will need to be updated as responsibilities change. Messages from senders not on the list will not be acknowledged, neither will their messages be distributed. Procedures have also been established for exchanging certificates with the IAEA Mailbox. Before any two individuals can communicate securely, there needs to be an exchange of certificates and an acceptance of those certificates. The acceptance process is normally completed by personal contact. The exchange of PKI certificates between an individual and the mailbox (being a non-person) requires special arrangements. In parallel to the Mailbox development, the IAEA is in the process of setting up a Public Key Infrastructure [4]; it should be fully operational later in 2007.

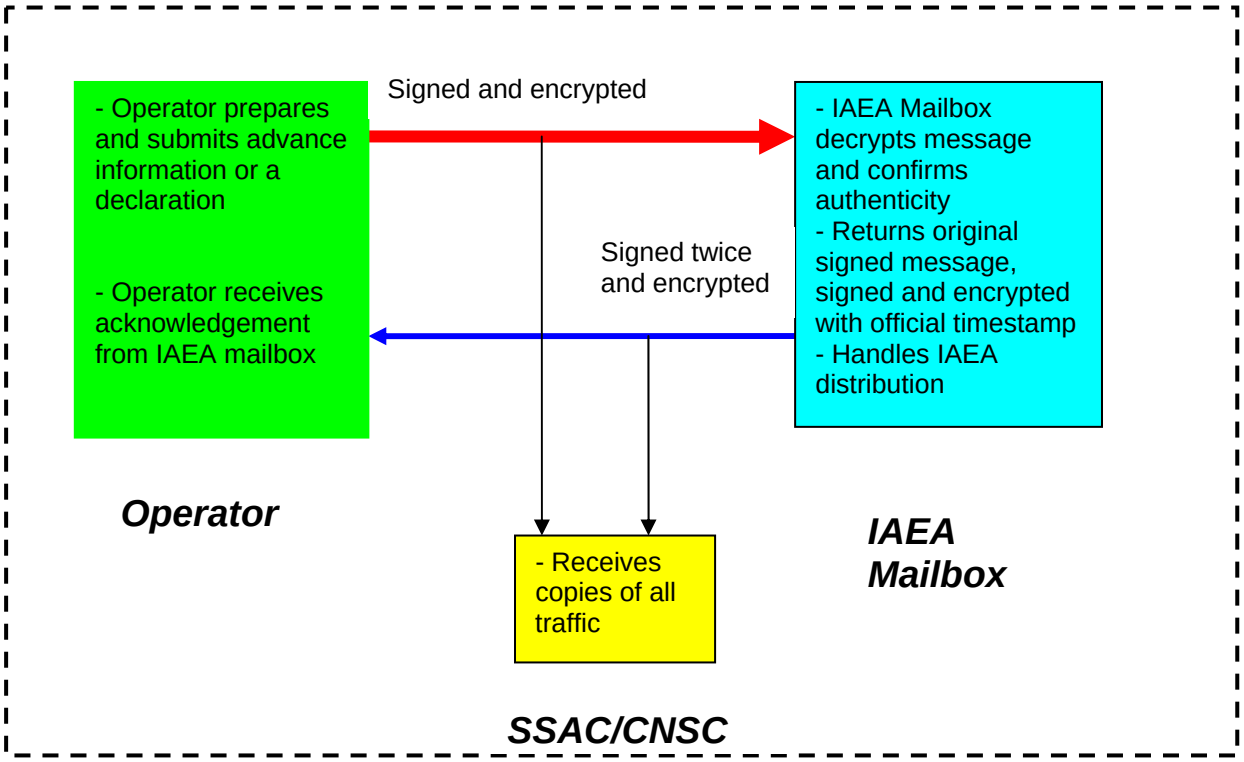



**Figure 3 – Exchange of information with the Mailbox**

When the message is received by the IAEA mailbox it is decrypted and its digital authenticity verified. If it is accepted, an acknowledgement is automatically sent to the originator and (if required) to the SSAC. The acknowledgement contains the original submission including the sender's digital authentication (signature) along with the digital signature and official time stamp provided by the IAEA Mailbox. The whole package is encrypted. Notice that the message contains two authentications within the encrypted package: the original signature of the sender and the time-stamped signature of the mailbox. This structure is automatically decoded by most off-the-shelf S/MIME software such as Outlook, however it has caused problems when using the Entrust Express plug-in for some versions of Outlook in that the body of the message and any attachments are not visible to the recipient. This problem is being addressed by Entrust, who will be issuing a patch[ii]. The two authentication blocks in the confirmation are necessary, so that:

- The originator can confirm that the original message has not been altered since the original signature is intact
- The time stamp issued by the IAEA cannot be altered since the IAEA signature is intact
- The sending of the original message by the originator cannot be denied by the originator since it contains his signature
- The IAEA's acknowledgement cannot be denied by the IAEA since it contains their signature
- The IAEA inspector can validate the original declaration while at the inspected facility by obtaining a copy of the message by any means and validating the corresponding signatures.

The official time for the delivery of the advance information will be that provided in the timestamp given in the Mailbox's acknowledgement. It will be a little later than that of the original transmission, but not significantly later. Should an operator not receive a timely acknowledgement (1-2 minutes), he should assume the message has not been delivered and he should consider the option of resending the message. The SSAC may retain an authentic copy of the original message sent by the operator and also the acknowledgement sent by the IAEA. If there is ever any dispute over the communication of advance information, the SSAC would have at its disposal the information to arbitrate.

## 6. Experience so far

The elements of the mailbox submissions and the timing of their transmission have been agreed upon. As of May 9, nine Unannounced Inspections have been successfully carried out at multi-unit stations. Advance information when used in conjunction with the Unannounced Inspection has provided a positive outcome in all cases. Based on current experience, the savings on the number of person days of inspection for transfers of fuel to dry storage has been of the order of 65%. The IAEA (electronic) Mailbox has been operational since late 2006, but the technical implementation at Canadian facilities has been delayed due to technical issues. When fully operational, the Agency will have rapid access to the most current information from each facility.

## 7. Conclusions

- The Unannounced Inspection and Mail box combination is now in operation
- This new verification tool is expected to save the IAEA considerable resources while retaining confidence in the inspection outcome
- The advance information to be provided to the IAEA has been agreed upon
- The technical arrangements for an electronic mailbox have been agreed upon
- For most commercial facilities, electronic submission complying with the IAEA's requirements can be achieved using standard desktop computers and software
- Full implementation of electronic submissions from Canadian multi-unit facilities is expected later in 2007.

## Acknowledgements

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[i] The use of the Word style editor within Microsoft Outlook must be avoided. It produces Microsoft specific formats which may not be supported elsewhere.

[ii] Entrust have promised to supply a patch for Entrust Express applicable to Outlook 2003 by May 11, 2007.

# SCORES – a Tool for Performance-based Safeguards Inspections

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***Abstract:***

The European Commission (EC) and the U.S. Department of Energy-National Nuclear Security Administration (DOE-NNSA) identified a common interest to develop a simple and practical tool for evaluating performance of nuclear safeguards. Such a tool would be aimed at identifying opportunities to improve safeguards system effectiveness and efficiency. The U.S. DOE has already implemented a simple assessment system in the field of nuclear material accounting and control. The system, which was developed at Lawrence Livermore National Laboratory (LLNL), is primarily based on compliance and performance testing but does not yet incorporate a full quantitative modeling approach. The U.S. DOE-NNSA and the EC plan to build on this initial effort in order to develop a quantitative model, *Standard Comprehensive Operational Rating Effectiveness System (SCORES)*, along with an associated software tool. This tool will enable site operators to conduct self-assessments and assist inspectors conducting verifications, with a focus on nuclear material accounting and control system performance. It is well suited for broader application. The SCORES approach could benefit other nuclear regulated areas and has the potential for non-safeguards applications.

**Keywords:** nuclear, safeguards, inspection, performance, assessment

## 1. Introduction

Performance assessment is one of the basic elements to achieve effective and efficient implementation of nuclear safeguards in the approaches of both the European Commission (EC) and the U.S. Department of Energy-National Nuclear Security Administration (DOE-NNSA). To facilitate such assessments, the EC and DOE-NNSA agreed to collaborate in developing a simple and practical tool for rapid evaluation of safeguards-relevant systems. The development effort will draw upon EC and US DOE experience in performing assessments of nuclear material accounting and control systems. Such assessments are primarily based on compliance or performance testing and will benefit from having a full quantitative modeling approach available.

The EC has begun developing methods for assessing capability maturity and obtaining assurance in the area of nuclear safeguards. This work follows a program outlined in newly revised approaches [3] for implementing Euratom safeguards. This is part of a drive to further even more the effectiveness and efficiency of Europe's supranational system of fissile material use supervision. Created alongside the European Atomic Energy Community in 1958, the Euratom control system relies primarily on the physical verifications conducted by its body of inspectors. This same body of inspectors performed the first multilateral nuclear inspection in 1960. Since then, the effectiveness of the Commission's controls has constantly increased. Its effectiveness is recognized in the cooperation agreements with the IAEA signed in 1973, 1976 and 1978 for the application of the international strengthened safeguards regime

in the European Community. Pioneer in the process leading to the Additional Protocols, the Community has accompanied the IAEA as the latter implements its integrated safeguards regime progressively across Europe. Alongside the European Union's enlargements, the advent of integrated safeguards is one of the reasons which prompted the European Commission to undertake a comprehensive review of its implementation approach for the – 'domestic' – Euratom safeguards system. This review process reached a milestone in early 2007 with the adoption of a text outlining a revised approach for nuclear safeguards in the Community. That text builds on shared understandings between the two European institutions competent on nuclear matters and requests the Commission to extend its safeguards 'toolbox'. One of the tools required is a methodology and a tool for assessing the performance of the operators' systems of accounting for and control of nuclear material. The EC expects to partly address this need through the joint development with the U.S. DOE of a novel aggregation model.

Lawrence Livermore National Laboratory (LLNL) has made initial efforts to develop pragmatic, rapid assessment tools for use in its international physical protection and material control and accounting programmatic work. These efforts have led to the development of a collection of metrics. Pilot tests using such metrics have been performed at selected sites in several countries. Many of these metrics were adapted from an insider vulnerability assessment software application and strategy database developed at LLNL [1]. These metrics provide an indication of safeguards *capability* with respect to one of the following topics: (a) physical protection, (b) material control and (c) material accounting. For each topic, the corresponding metric (e.g. for an inventory metric in material accounting, illustrated in Table 1) is graduated from weaker to stronger strengths of the relevant safeguards capability. Each level in the graduation is associated to a description of the expected safeguards capability context.

| Level | Safeguards Capability Context Description |
|---|---|
| 1 | No mechanism is currently in place to detect discrepancies between inventory records and actual material quantities present in a given location. This may be because:<br>i) Inventories are not conducted<br>ii) Inventories are done, but the results are not compared with records for discrepancies<br>iii) Records never kept |
| 2 | Inventories done but the implementation has one or more of the following flaws:<br>i) An insufficient percentage of the inventory is sampled when inventories are conducted<br>ii) The time between inventories is inappropriate. A discrepancy between records and actual amount of material present may not be detected for a long time |
| 3 | Frequency of inventories as well as percentage of inventory sampled is appropriate (e.g., follows a relevant guideline from IAEA or DOE), but has one or more of the following flaws:<br>i) Inventories are not conducted by qualified personnel properly trained in inventory procedures<br>ii) Critical steps under total control of one person. For example, checking for item presence, or recording, or transmittal of inventory results done by one person |
| 4 | Effective inventory system with all critical elements present and no major implementation flaws. Partial and random inventories conducted frequently, full inventories conducted occasionally (or once initially followed by systematic sampling)<br>The personnel conducting the inventories are properly trained, no critical step is under total control of just one person and the process would be difficult to manipulate |

**Table 1:** Example Inventory metric in the area of Material Accounting

Each metric provides insights by using observables such as implementation flaws, or limitations related to procedures or equipments. By identifying the level of a given safeguards capability in a facility, site teams can better track changes impacting that capability, monitor their implementation, help avoiding flaws, and plan future improvements.

While the insights provided by individual metrics are valuable, other important insights can be gained from aggregated views of the metrics for a facility. In existing work, LLNL developed a quantitative model which aggregates metric levels and gives a higher level view of safeguards and security capabilities in a facility. Using this model LLNL could make capability assessments with respect to three broad topics: **outsider threat**, **insider threat**, and **material accounting** (providing assurance and after-the-fact detection).

More specifically, the LLNL pilot aggregation model takes the metric level "assignments" for a facility as an input and produces three summary numbers reflecting the facility's performance relative to the three broad topic above. Specifically, these numbers reflect how a facility would marshal capability resources to address the outsider threat, insider threat, and material accounting concerns, respectively – with the help of a respective sponsoring program, if applicable. Furthermore, the LLNL model also computes sub-aggregations numbers, when going from the metric levels to the total three summary numbers. These numbers reflect sub-concerns of intermediate level. Thus, the model can also be used to trace reasons for the summary numbers in terms of judgments about the metric levels for the facility. This approach was successfully used to monitor progress of international programmatic activities sponsored by US DOE-NNSA on material control & accounting and physical protection.

Security systems aspects such as physical protection fall outside the scope of the collaboration between US DOE-NNSA and the EC on the development of SCORES. The model joint development will focus on safeguards capabilities only.

The US DOE-NNSA and the EC plan to build on the initial effort described above in order to develop a novel quantitative model, named *Standard Comprehensive Operational Rating Effectiveness System (SCORES)* along with an associated software tool. This tool will enable facility operators to conduct self-assessments and assist inspectors conducting verifications. It will facilitate assessments of safeguards capabilities with a focus on nuclear material accounting and control system performance.

Interpreting the SCORES aggregation model results for a given facility will help addressing questions such as:

- How have safeguards capabilities increased when the facility capabilities' *current* levels are compared to its *baseline* levels – e.g. identified prior to an improvement program, or agreed via an audit?
- How will the capabilities further increase when all *safeguard improvement* activities are complete?
- Do the *goals* for a facility represent a pragmatic (sensible) stopping point in terms of capabilities?

The next section illustrates the basic concepts of the existing LLNL approach for aggregating metric levels into a summary number. This uses a subset of the US DOE-NNSA material protection control and accounting (MPC&A) metrics. The third section describes a representative pattern of results from applying LLNL's aggregation model in pilot applications. No specific facility is referenced, but insights and interpretations in this section reflect actual results obtained by US DOE-NNSA. The final section summarizes the paper and highlights expected benefits of the SCORES aggregation model/tool.

## 2. Basic concepts and use of the LLNL aggregation model

Intermediate and summary numbers produced by the LLNL aggregation model are scaled to fall between the following limits. A facility assigned the weakest levels (i.e., level 1) on all metrics obtains an overall result of 0. A facility assigned the strongest levels on all metrics obtains an overall result of 1. Facilities with metrics level assignments in between the weakest and strongest obtain results in between 0 and 1. Once computations are complete, numerical results can be multiplied by a convenient constant (e.g., 100) for presentation purposes.

The numbers in the model can be interpreted as performance indicators. In general, they are arranged in a hierarchy designed to aggregate lower level metrics into higher level indicators. For example, a Material Accounting performance indicator might be constructed by aggregating of the following four metrics:

Material Accounting
1) Inventories
2) Measurements
3) Information management
4) Tamper indicating devices (TID)

The highest level summary numbers are computed using numbers from the level below. In turn these are computed from levels still lower and so on in a "drilling down" process. The whole computation is ultimately driven by the individual metrics. At each sub-level, the computation process gives intermediate summary numbers representing sub-concerns. The process towards computing more and more refined indicators continues until each sub-concern actually coincides with the topic of an individual metric. The numbers obtained at any level in the hierarchy with this aggregation model can be formally related to "archetype" safeguards capability contexts. This relation is established by assigning context descriptions to typical number ranges. There is some degree of flexibility in establishing these relations. In the generic case, the ranges and corresponding context descriptions proposed in the LLNL pilot work are shown in Table 2.

| Number Range | Safeguards Capability Context Description |
|---|---|
| 0 – 20 | negligible capability in this area |
| 21 – 40 | severely limited capability (some infrastructure present, but major capability gaps) |
| 41 – 60 | limited capability (infrastructure present but significant problems still exist) |
| 61 – 80 | significant capability (infrastructure present but a few concerns still exist) |
| 81 – 100 | in-depth capability (in-depth infrastructure with, at most, relatively minor concerns) |

**Table 2:** Generic Interpretations of Aggregation Model Numeric Ranges

Overall, different combinations of metric levels can result in similar scores for a given indicator. The LLNL aggregation model has been formulated so that facilities with the similar scores for a given indicator can be compared. In such a case, the facilities are considered to have equivalent "mobilizing of safeguards capabilities". In other words, they show equivalent ability to address the given concern, at the level of detail of the model's descriptions[1]. Back to the example, two facilities can obtain similar scores indicating an equivalent ability to address material accounting concerns, and one achieves this by using a strong information management system, whilst the other achieves it through good inventory taking procedures.

The LLNL aggregation model allows for creating a concept of "benchmark". This means a reference of progress on safeguards capabilities, applicable across facilities. It is created as follows. For each metric, levels are selected depicting a hypothetical benchmark facility. These levels represent *pragmatic* capability progress per topic, for all metrics. In almost all cases, such levels will not be the highest of each metric scale. Rather, they will represent capability progress reasonably achievable for many facilities, yet providing significant safeguards capability levels.

[1] The LLNL metrics and aggregation model do not purport to do quantitative vulnerability assessments. Such assessments calculate probabilities of a facility win against specific adversary types using specific safeguards defeat methods. The LLNL metrics and model described here are rather tailored to reflect, at a pragmatic level of detail, how much progress in safeguards capabilities a facility achieves for the summary areas of concern. A similar observation applies for the model to be jointly developed between the EC and the US DOE-NNSA.

Analyzing this hypothetical benchmark facility with the LLNL aggregation model, a family of summary number will result. In the example above, these scores would describe the hypothetical benchmark facility's ability to address material accounting concerns.

The model may be applied to compare an actual facility to the benchmark. The summary numbers computed from that facility's metric levels should be compared to the benchmark summary numbers. If the figures computed for the facility meet or exceed the benchmark figures, this indicates that the facility has progressed to a level of ability equivalent to the benchmark or exceeding it. Such a facility need not have individual metric levels comparable to that of the benchmark. The facility may have higher metric levels for some topics, and lower levels for other topics.

Applying the interpretations in Table 2 to the material accounting indicator example discussed above helps understanding how to use the aggregation model for tracking progress in a facility's safeguards capabilities. The aggregation model can show how the facility proceeds through successive stages of improvement:

a) **baseline** – the metrics levels identified for the facility, e.g. via a reference audit, or prior to the beginning of a specific improvement program;
b) **current** – the metrics levels currently observed at the facility;
c) **committed** – the metrics levels expected to be attained once work that has already been committed to (e.g., budgeted or begun) is completed
d) **goal** – metrics levels deemed appropriate for the facility to attain, on the basis of a cost versus benefit analysis of further efforts and of the *endpoint* capability objectives

As the facility progresses through these stages, the summary numbers computed using the model give insight on the overall progress. They reflect the evolution of the level combinations of the individual metrics. The teams in charge can thus both monitor current work and plan or revise future work to help achieving the most meaningful progress given resource and programmatic constraints.

In the LLNL model, two quantitative procedures are used to aggregate the lowest-level metrics. In the first procedure, the model associates function to each metric. This function may be a discrete or continuous. It permits assigning a numerical value to each level of the metric. In the second procedure, weights are used for combining the numerical values of each metric into a single number. These weights are chosen upstream when developing the model. A fair portion of the SCORES joint development plan focuses on this aspect.

The aggregation scheme may be a simple weighted sum, or a more complex structure. In the LLNL model, for instance, multiplicative structures are also used. Models using multiplicative structures can be calibrated to exhibit complementary or compensatory interactions among the metrics [2]. Complementary interactions are appropriate in situations where a high score on several different metrics is needed to achieve a high aggregate score. The material accounting indicator example above can be used to illustrate this situation. The LLNL model heuristically expects less progress overall on material accounting performance if improvement focuses on one single metric instead of balancing improvements across the four topics underpinning the indicator. In this example, the LLNL model assumes complementary interaction between the four metrics. This is appropriate when a weakness on one metric tends to "negate" the value of progress on the other metric. Earlier analyses when developing the LLNL model concluded that for material accounting, typically, all four topics need to work together for an effective capability. Compensatory interactions refer to situations where a deficiency in one area can be mitigated by a high score in another area. This can happen for instance when modeling the performance of surveillance systems or other systems involving detectors. For example, a weakness in one type of sensor (position sensors) can be compensated for by strength in another type of sensor (area intrusion sensors) and vice-versa.

## 3. Representative results from applying aggregation model

### 3.1. High level results, summary numbers and progress tracking

This section illustrates the kinds of insight provided by the LLNL aggregation model. The figures are a snapshot taken from an actual facility example. The purpose is to illustrate high level summary results obtained by US DOE-NNSA when applying the LLNL model (see Table 3).

As explained above, security systems' aspects are beyond the scope of the SCORES model joint development. In this example, the full LLNL pilot model for MPC&A capabilities is illustrated, including performance metrics of security systems related to both insider and outsider threats. This is required to illustrate how performance for multiple high level concerns can be analyzed.

| Facility XYZ | Material Accounting | Insider Threat[2] | Outsider Threat[2] |
|---|---|---|---|
| Baseline | **8** | **21** (24) | **19** (26) |
| Current | **53** | **40** (46) | **31** (47) |
| Committed | **71** | **83** (92) | **70** (71) |
| Goal | **71** | **83** (92) | **74** (76) |

**Table 3:** Illustrative Aggregation Model High Level Summary Results

The high level summary results in Table 3 address the three questions mentioned in the introduction and represent a common pattern seen by DOE-NNSA in pilot applications, namely: how much capability increase is being observed, how much increase is expected from actions committed to and whether the agreed capability goals are achievable.

*How has capability increased at a facility considering the baseline status of the facility – i.e. before improvement activities began – and the facility's current capabilities?* In this example of facility XYZ, the *baseline* numbers reflect contexts where safeguards capabilities are deemed close to negligible for all three high level areas of concern. The *current* capabilities at the moment of the snapshot represented meaningful and substantial progress in improving the safeguards infrastructure, but still reflected some major gaps. In general, current capabilities depend on how much work has actually been completed at a facility.

*How will capability increase when all committed improvement activities are complete?* The *committed stage* numbers for all three high level areas of concern are considerably higher than the "current" numbers. They represent capability contexts where no significant problems or major gaps are present, although a few concerns may remain about the outsider threat and material accounting. This is characteristic of a facility where the actual work, though committed to, is not yet complete.

*Do the goals for a facility represent a pragmatic (sensible) stopping point in terms of capabilities?* For the improvement program illustrated in this example a benchmark had been set at 70 for all three high level areas of concern. For facility XYZ, all three *goal stage* numbers exceed this benchmark reference number of 70. This represented significant capability contexts for the outsider threat and material accounting, and an in-depth capability context for the insider threat. The committed stage and goal stage scores in this example are similar for all three high level areas of concern. This illustrates a situation where most improvements have already been initiated. In such a case, very little additional "new activities" should be required.

Figure 1 displays the results of the aggregation model in a chart form. The evolution of triangular shape depicts the evolution of the facility indicators along the three high level capability areas simultaneously. For facilities where planning is just beginning, it would not be unusual for such a chart to depict identical profiles for the 'baseline', 'current' and 'committed' stages, with a 'goal' profile much bigger. Then, as work progresses, the shape for the 'current' and 'committed' stages would become intermediate between the 'goal' and 'baseline' shapes. If all goal activities are successfully initiated,

[2] In this example, the numbers in parentheses for the Insider threat represent the summary disregarding general separation of personnel duties aspects; those in parentheses for the outsider threat represent the summary disregarding response force aspects. These parenthetical numbers provide insight for situations where it may not be within the purview of the sponsored program to help limit duties and authorities for particular personnel (insider) or help upgrade aspects of the response force (outsider) at a facility.

the 'committed' and 'goal' shapes would become the same. At the end, the 'current' shape would reach the 'goal' shape as all work is completed.

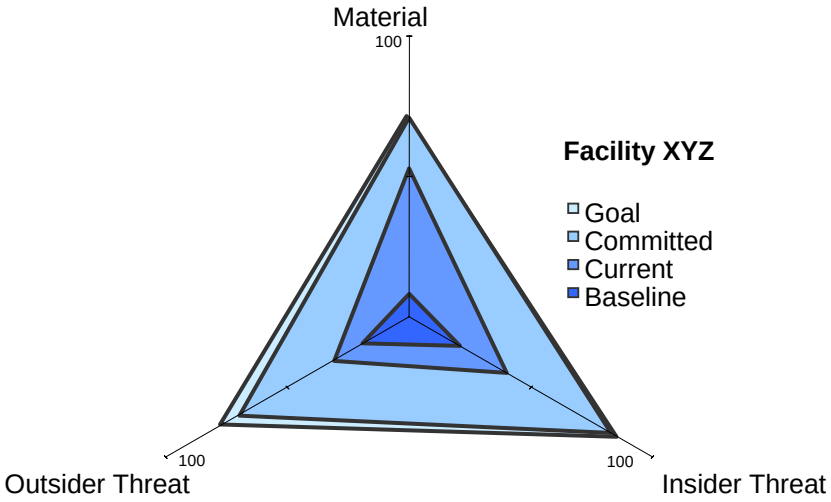


**Figure 1:** Illustrative Aggregation Model High Level Summary Results

The next subsection illustrate how "drilling down" to the next level of detail for material accounting gives insight into the reasons for the summary number results.

### 3.2. Material accounting drill down

The figures in Table 4 can be used for understanding more about how the composite performance score related to the material accounting indicator for facility XYZ was obtained. The figures in the column called Material Accounting were actually computed from the rightmost four columns, reflecting the complementary interactions between these metrics discussed above.

| Facility XYZ | Material Accounting | Inventory | Measurements | Information Management | TID[3] |
|---|---|---|---|---|---|
| Baseline | **8** | **0** | **0** | **50** | **10** |
| Current | **53** | **100** | **50** | **50** | **70** |
| Committed | **71** | **100** | **75** | **50** | **100** |
| Goal | **71** | **100** | **75** | **50** | **100** |

**Table 4:** Illustrative Material Accounting Results

The table shows that in going from the baseline to the current stage, substantial improvements occurred simultaneously in inventory, measurements, and TID-related safeguards. This was reflected in the evolution of the material accounting capability indicator. It improved from a score of 8 (meaning "negligible capability") to a score of 53 (the equivalent of "significant problems remain"). At the moment of the snapshot, the improvements in the measurements area and on the use of tamper indicating devices slated for the committed stage actually brought the material accounting indicator above the benchmark score of 70 points.

Improvements in information management systems were not needed for the facility XYZ to reach the benchmark goal for material accounting. This illustrates how the aggregation model can be used to examine which combination of metrics improvements will attain a benchmark goal, and which will fall short.

[3] Tamper Indicating Device

## 4. Summary and conclusions

The EC and the U.S. DOE-NNSA share interests in the development of a complete methodology for the assessment of safeguards-related systems' performance. Building a tool that helps both nuclear operators and control bodies – e.g. inspectorates – in applying the methodology alike is perceived as a benefit by both organizations. Utilizing the existing agreement for cooperation on safeguards R&D, the two organizations agreed to launch a collaboration which puts together the experience of LLNL in the development and testing of a pilot aggregation model and the recent work undertaken by the EC on methods for the assessment of safeguards capability maturity and guidelines.

This scope of the collaboration on SCORES does not extend to security systems. Nevertheless, the resulting model is expected to have features which may interest other nuclear regulated areas and could have potential for non-safeguards applications elsewhere. Beyond the partners in the collaboration, the work developed has potential for being of interest both to national inspectorates and to other organizations such as the IAEA.

The LLNL experience shows that an aggregation model and analysis methodology can provide a simple and pragmatic tool for evaluating safeguards capabilities at monitored facilities. The scoring functions used to implement the aggregation can be configured to represent a rich set of relationships among the individual metrics, beyond simple weighting and summing.

The example discussed above illustrates how aggregation models such as the one developed by LLNL can:

- analyse a facility's safeguards capabilities and progress made, gaining insights from a high level perspective;
- allow both facility management and nuclear inspectors to:
  - focus on areas of concern where identified weaknesses prevail;
  - initiate additional targeted improvement projects – e.g. focussing on areas where the "commitment" stage result is far from the "goal" stage result;
  - monitor current and expected progress about the facility's safeguards capabilities, – e.g. using the results from the "drill-down" mechanism;
  - define a sensible stopping point when optimum efficiency has been achieved and compare it to a benchmark
- let nuclear control bodies gain insights about safeguards capabilities and performance levels across a population of facilities – e.g. grouping together facilities with comparable ability levels.

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# Development of State-Level Integrated Safeguards Approaches for Countries with Developed Nuclear Activities

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***Abstract:***

In States having concluded a protocol additional[1] to their comprehensive safeguards agreement[2], the Agency is permitted to perform additional activities and is able to undertake a more thorough assessment of a state's nuclear activities through analysis of additional information provided by the state. The intention of such work is to allow the Agency to confirm that, in addition to the absence of diversion of declared material or misuse of declared facilities, no undeclared nuclear material or activities exist within a state. This is known as the 'broader conclusion'.

Once such a conclusion has been drawn, the level of safeguards activities for the verification of declared nuclear materials may be reduced to a certain extent, particularly in connection with less sensitive nuclear materials (e.g. depleted, natural and low enriched uranium and irradiated fuel) and facilities such as power reactors, irradiated fuel storages and low enriched fuel fabrication plants. The combination of safeguards activities undertaken in such a case is referred to as 'Integrated Safeguards' as it includes aspects of traditional and strengthened safeguards based on consideration of the State as a whole.

The particular approach to safeguards activities under integrated safeguards is significantly dependent upon the extent of the nuclear fuel cycle activities within the state in question. This paper discusses the pre-requisites necessary for the introduction of integrated safeguards within a State, the objectives of an integrated safeguards approach and some of the concepts considered in developing integrated safeguards approaches in countries with developed nuclear activities.

**Keywords:** integrated, safeguards, approaches, state-level

## 1. Introduction

Integrated Safeguards is not simply the summation of traditional safeguards activities and additional measures available under the additional protocol[3-5], rather, it has been defined as 'the optimum combination of all safeguards measures available to the Agency under comprehensive safeguards agreements and Additional Protocols which achieves the maximum effectiveness and efficiency within available resources in fulfilling the Agency's right and obligation in paragraph 2 of INFCIRC/153 (Corrected)'[6].

The basic principles governing the development of integrated safeguards are that they should:

- be non-discriminatory, i.e. that the same technical objectives should be pursued in all states with comparable safeguards obligations, although the measures actually used in individual states may differ;
- be based on State-specific considerations gained from a comprehensive information evaluation of a state;
- take account of all plausible acquisition paths by which a State might seek to obtain nuclear material for a nuclear explosive device; and

- be founded on the premise that nuclear material accountancy should remain a safeguards measure of fundamental importance.

Additionally, overall cost neutrality in respect of safeguards implementation remains a goal in developing integrated safeguards approaches. Whilst the costs of implementing safeguards in a State may be expected to rise immediately following the entry into force of an Additional Protocol due to the increased activities necessary to verify a State's declarations under the Additional Protocol and to check the consistency of all information available to the Agency, it should be expected that over the medium term the costs would reduce to the level under traditional safeguards but with an increased level of effectiveness. In order for this to occur, certain reductions in inspection effort relating to less sensitive facilities and materials may be applied under integrated safeguards, for example the timeliness goal for irradiated fuel may be increased from three to twelve months under integrated safeguards. The rationale for such relaxations is that activities performed under a comprehensive safeguards agreement alone are predicated on the assumption that undeclared nuclear activities e.g., undeclared facilities for reprocessing or enrichment may exist undetected whereas under a comprehensive safeguards agreement supported by an additional protocol, the Agency's ability to provide credible assurance of the absence of such undeclared facilities changes this assumption and therefore creates the potential for changes in implementation parameters and reductions in verification effort for declared nuclear material[7]. Such relaxations are used in offsetting increased or alternative activities at headquarters and in the field including information review and analysis, complementary access and the use of more advanced technology and remote monitoring.

## 2. Drawing 'the broader conclusion'

Integrated safeguards measures may only be applied once it has been concluded that, in addition to all declared nuclear material in a state being properly accounted for, there exists no undeclared nuclear materials or activities.

Of course, a negative hypothesis such as this cannot be proved with absolute assurance and thus the conclusion is drawn based on inference from the absence of any information to the contrary from a thorough investigation of relevant information. Thus the basis for drawing the broader conclusion differs fundamentally from more traditional safeguards activities and specifically conclusions drawn concerning the absence of diversion of declared nuclear material. Assessments concerning declared nuclear materials are made primarily on the basis of quantitative measurements concerning a defined data set which includes information on the amount and form of nuclear material and facility design information provided by a state. Verification activities may then be performed to ensure that conclusions can be drawn with defined statistical reliability and uncertainty. On the contrary, a conclusion of the absence of undeclared nuclear material and activities is based largely on qualitative assessment of wide range of information including that provided by the state, determined directly by the Agency through safeguards activities and obtained from other sources.

In general terms, the following conditions can be considered as pre-requisites to the drawing of the broader conclusion:

- the state has complied with the requirements of its safeguards agreement and additional protocol;
- measures for the verification of declared nuclear activities have been implemented by the Agency and no indications of the diversion of nuclear material have been found;
- a comprehensive state evaluation based on all information has been performed;
- issues identified by the state evaluation process have been addressed, e.g. by, seeking clarifications from the state; and,
- the absence of undeclared nuclear materials or activities has been confirmed at sites, mines, concentration plants etc. e.g. by performing complementary access under the Additional Protocol.

If, following the implementation of integrated safeguards, the Agency was not able to reaffirm the conclusion of the absence of undeclared nuclear material and activities for a state as a whole, corrective actions would have to be taken which, depending on the circumstances, could include increasing traditional safeguards activities in the state, whilst continuing to implement the measures of

the Additional Protocol. It is therefore imperative that the safeguards activities performed under an integrated safeguards approach are sufficiently extensive to enable the reaffirmation of the broader conclusion to be made in a robust manner.

In practice, integrated safeguards consist of activities performed under a comprehensive safeguards agreement being combined in an optimal manner with activities under an Additional Protocol. This optimal combination is based, *inter alia*, on consideration of the features of a state's nuclear fuel cycle and the results of information analysis which result in what is termed the 'state-level approach'. These factors are considered in the following section.

## 3. Integrated safeguards approaches in states with developed nuclear activities

In developing a State-level integrated safeguards approach (SLA) for countries with developed nuclear fuel cycles, advantage can be taken of the experience gained by the Agency in developing integrated safeguards approaches for specific facility types and for groups of facilities e.g. at a site. The tools developed to support these more specific approaches, can be used as fundamental building blocks when developing an approach intended to cover a state as a whole. Such tools include Unannounced Inspections or Short Notice Random Inspections, the use of more detailed information and advanced declarations by the State and an approach based on functional sectors.

Development of an SLA for a developed nuclear State may therefore include the following activities:

- determination of State specific characteristics e.g. nuclear programme, related R&D and relevant infrastructure;
- acquisition path analysis on a State level and determination of credible diversion and concealment strategies; this should also cover the potential collaboration between states and non-state entities;
- consideration of consistency of R&D activities and facilities to the declared nuclear programme;
- establishment of State-specific safeguards measures;
- determination of functionally related facilities and nuclear material flows;
- determination of functionally related R&D activities;
- development of activities to be performed under the SLA based on consideration of generic and State-specific safeguards issues; and,
- development of an evaluation methodology.

Key to this approach is the analysis of functional relationships and the definition of appropriate functional sectors, the idea being to verify the fuel cycle as a whole rather than as isolated facilities. These sectors are defined so as to allow simultaneous consideration of related facility types, nuclear material types, nuclear material flows and R&D activities. The requirement for advanced notification by the state of nuclear material flow is an effective means of allowing routine interim inspections to be replaced by randomised inspections, thereby increasing efficiency whilst improving the effectiveness of safeguards.

Under an SLA, the overall safeguards objectives continue to be[8]:

- timely detection of diversion of declared nuclear material,
- detection of undeclared production or processing of nuclear material at declared facilities,
- detection of undeclared nuclear material and activities in a State as a whole.

SLAs developed using the approach outlined above should allow the fulfilment of these objectives in the most cost effective manner. It is also imperative that SLAs offer a high level of effectiveness and act as a credible deterrent against any proliferative forces which may exist in a State with a well developed nuclear infrastructure.

### 3.1 Unannounced or short notice inspections

The use of unannounced inspections i.e. inspections with only minimal advanced notification regarding inspection timing, activities and location being provided to a State are an important component of integrated safeguards approaches. Due to their unpredictability by the State or facility operator, unannounced inspections contribute not only to the detection of diversion and the misuse of a facility but also to the deterrence of such activities. A properly executed randomised approach to a group of facilities allows significant gains to be made in cost effectiveness without a concomitant reduction in safeguards effectiveness as, whilst the number of facilities and amount of nuclear material actually inspected is reduced, the specific inspections are not known in advance to the operator or State and there is thus an added level of deterrence which applies to all facilities subject to the randomised approach, inspected or not.

### 3.2 Containment and surveillance measures (C/S) and remote monitoring

C/S has played an important role in many safeguards activities over many years and will continue to do so in the future. The additional protocol includes provisions that facilitate the use of technological advances in safeguards through transmission of data from unattended containment/surveillance or measurement devices (Article 14.a). As will be discussed below, when considering the integration of safeguards activities through an entire State, analysis of material flows from facility to facility are particularly important and the potential for using remote monitoring and sensing devices towards this end is particularly attractive in terms of its positive potential impact on efficiency without compromising effectiveness.

Additionally, advanced technology may be effectively used in achieving safeguards objectives where difficulties exist in effectively performing unannounced inspections. Alternatively unattended or remote monitoring can be used to complement a reduced number of randomised inspections as in such cases the physical presence of inspectors at a facility is not required for safeguards relevant data to be obtained or analysed.

### 3.3 Facility-specific approaches

As mentioned above, under integrated safeguards activities concerning certain less sensitive facilities and materials may be reduced comparatively to those necessary in the absence of the broader conclusion.

The principles governing work on facility-specific integrated safeguards approaches include:

- provision of coverage for all plausible diversion and misuse scenarios associated with the specific facility type;
- retention of nuclear material accountancy as a fundamentally important safeguards measure;
- annual material balance verification, using random selection of facilities where appropriate; and,
- non-discrimination among States where integrated safeguards are applied by application of the same safeguards objectives to all facilities of a specific types.

Take for example the most common type of nuclear facility throughout the world i.e. the uranium dioxide fuelled light water reactor (LWR). Such facilities account for some twenty percent of the Agency's total inspection effort and involve only less sensitive nuclear material. Integrated safeguards approaches in respect of LWRs have as their technical objective the ability to detect and deter the diversion of significant quantities of nuclear material with a twelve months timeliness verification goal and to be able to detect and deter the misuse of the facility (e.g., undeclared production of direct use material or fuel pin exchange or removal) and borrowing of fuel assemblies from other facilities to conceal diversion of nuclear material.

Reduction in safeguards activities such as elimination of surveillance between physical inventory verifications requires certain conditions to have been met. In addition to the generic issue of the broader conclusion, in all considerations relevant to the application of integrated safeguards, there are facility-type specific factors such as:

- agreement with the state/facility operator for the advance notification concerning shipments and receipts of fuel assemblies and the operational programme of reactors;

- the ability of the Agency to verify at LWRs any shipments of irradiated fuel to be stored under difficult to access conditions;
- arrangements for the practical implementation of effective unannounced inspections have been made.

Where these are met, the basic features of integrated safeguards as applied to $UO_2$ fuelled LWRs are:

- an annual PIV at each reactor refuelled in the year;
- a PIV equivalent at a randomly selected proportion (50%) of reactors not refuelled in the year;
- a reduced number of randomly selected interim inspections at a proportion of reactors; and,
- the sealing of the reactor core between refuelings.

Other facility-specific approaches for facilities handling only similarly low sensitive materials have been developed. Again, these allow a reduction in safeguards inspections without a corresponding reduction in effectiveness.

### 3.4 State-specific features

The integrated safeguards approaches for particular states need to take account of state-specific features and characteristics in optimising safeguards effectiveness and efficiency. This is particularly important for states with more developed nuclear activities. Factors to be considered include:

- the structure and components of the nuclear fuel cycle from uranium mines to nuclear waste;
- the number and types of nuclear facilities and locations outside of facilities and the associated activities conducted on nuclear sites;
- the inventory and flow of nuclear material within and between facilities;
- fuel cycle-related research and development;
- the manufacture and trade of sensitive nuclear-related equipment and materials; and
- correlations between the above features.

### 3.5 Site approaches

In some respects integration of safeguard activities at the State level can be considered as an extension of such integrations at the site level particularly in the case of a site containing several different facility types.

Consider for example a hypothetical nuclear site containing the following facilities:

- an LWR power park containing a number of $UO_2$ fuelled reactors;
- irradiated fuel storage;
- reprocessing plant;
- $UO_2$ and MOX fuel fabrication plants;
- R&D facilities including hot-cells for post-irradiation examination.

Analysis of these facilities shows that certain nuclear material flows are to be expected between facilities on the site e.g.

- fresh fuel from the fuel fabrication plants to the reactors;
- irradiated fuel movements between reactors, storage and reprocessing plant;
- uranium and plutonium products from the reprocessing plant to the fuel fabrication plants;

One means of considering these material flows is to consider the facilities concerned not as individual entities but as one super-facility which can then be subdivided on the basis not of physical identity but in terms of the nuclear material inventories and flows. Each individual component so determined may then be referred to as a sector. The example site described above might be sub-divided as so:

Sector a: $UO_2$ fuel fabrication plant (assembly area), fresh fuel storage at reactors and $UO_2$ product storage at reprocessing plant;
Sector b: LWR reactors and irradiated fuel storage at reactor, storage facility and reprocessing plant;
Sector c: reprocessing plant;

Sector d: MOX fuel fabrication plant and $PuO_2$ storage area at reprocessing plant (depending on the extent of the process performed at the fabrication plant this sector may be extended to included plutonium nitrate intermediate storage and process areas at the reprocessing plant);
Sector e: $UO_2$ fuel fabrication plant (powder area) and $UO_2$ powder storage area at reprocessing plan; and,
Sector f: R&D facilities which are outside of the routine sequential flow of materials but which might receive or ship material to or from the reactors, reprocessing plant or fuel fabrication plants.

Given this breakdown structure, the potential exists for a reduction in certain safeguards activities concerning sequential nuclear material flows from one facility to another as the export from one facility is the import to the next and as such, as long as continuity of knowledge of the material is retained, there is need only to verify the shipment or receipt. Consideration of the physically separate facilities as a virtual super-facility also provides an element of deterrence against misuse through the use of randomised short notice sector-based inspections. An important pre-requisite to this approach is the frequent advanced declaration of inventories and flows such that even given the short notification aspect, it should be clear to the Agency how much of each material type (strata) should be expected in a particular sector chosen for verification.

Additionally, the consideration of sectors based on material type is an effective means of addressing borrowing issues which are particularly significant in the case of complex sites with interrelated facilities, the transfer of material between which is relatively straightforward.

### 3.6 The state as a whole

The above arguments can of course be extended to all facilities in a particular state. Consider the nuclear material flows indicated in Figure 1 between different facility types which may or not be on separate sites.

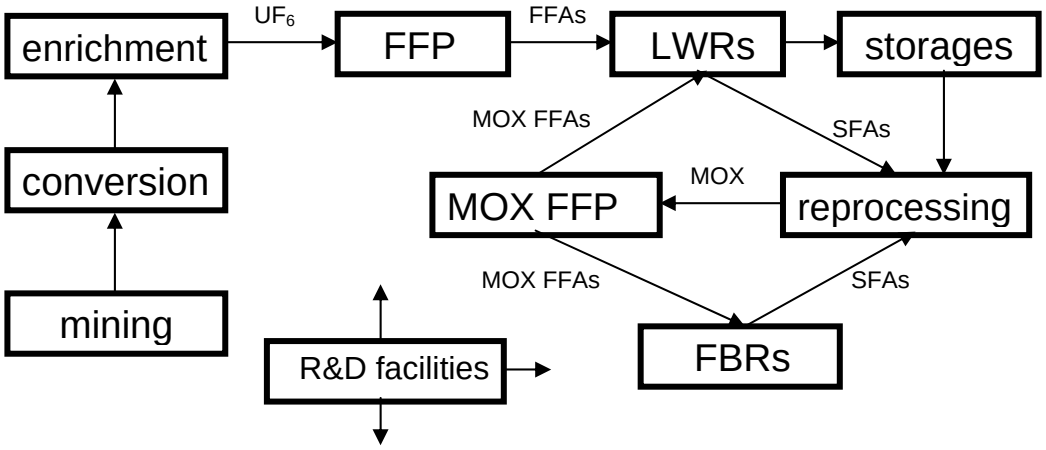


Figure 1. Relationship diagram of nuclear fuel cycle facilities

- $UF_6$ may be expected at conversion, enrichment and fuel fabrication plants (FFPs);
- unirradiated uranium dioxide fuel assemblies may be expected at FFPs and at LWRs;
- unirradiated MOX fuel assemblies may be expected at MOX FFPs, LWRs and fast breeder reactors (FBRs)
- irradiated fuel assemblies may be expected at LWRs, irradiated fuel storages and reprocessing plants;
- plutonium powder may be expected at reprocessing plants and MOX fuel fabrication plants as might plutonium nitrate solution depending upon the extent of activities performed at the fabrication plant;
- smaller amounts of all-types of nuclear material may be expected at research and development facilities.

Therefore it is seen that when considering the activities to be performed under an integrated safeguards regime, it is important not to simply consider particular facility types in isolation or even in combination on the basis of co-location but rather to examine all inter-relationships between facilities and facility types within a state.

In addition to the facilities at the hypothetical site described above in section 3.6, the case depicted in Figure 1 indicates additional interactions between facilities when the state is considered as a whole, as in this case more facilities must be considered, these include:

- additional LWRs;
- conversion facility;
- enrichment facility;
- a further $UO_2$ fuel fabrication plant; and,
- additional R&D activities.

With these additional facilities, the division of sectors described above may be extended, one possible description being:

Sector i: mining and conversion;
Sector ii: enrichment and conversion of $UF_6$ to $UO_2$
Sector iii: $UO_2$ fuel fabrication plant, fresh fuel storage at reactors and $UO_2$ product storage at reprocessing plant;
Sector iv: $UO_2$ fuelled LWR reactors and irradiated fuel storage at reactor, storage facility and reprocessing plant;
Sector v: MOX fuelled LWRs, FBR and irradiated fuel storage at reactor, storage facility and reprocessing plant;
Sector vi: reprocessing plant;
Sector vii: MOX fuel fabrication plant and $PuO_2$ storage area at reprocessing plant (depending on the extent of the process performed at the fabrication plant this sector may be extended to included plutonium nitrate intermediate storage and process areas at the reprocessing plant); and,
Sector viii: R&D facilities which are outside of the routine sequential flow of materials but which might receive or ship material to or from the reactors, reprocessing plant or fuel fabrication plants.

Such a conceptual breakdown may be used to include all nuclear fuel cycle related activities and to enable safeguards activities to be planned and conducted in an efficient manner analogously to, and by extension of, the site-based approach described above.

Lastly, the role of complementary access should be mentioned as this is an essential part of a state-level integrated safeguards approach. The short notification aspect of CA adds to the deterrence against misuse of facilities or materials provided for by other short notice inspections adopted as part of integrated safeguards approaches particularly as the range of locations subject to CA is somewhat wider than that for other short notice inspections. CA may also be used in resolving questions or inconsistencies related to a state's declaration or other information and can be particularly effective in ensuring that undeclared activities are not being performed at sites and facilities not usually subject to routine safeguards activities.

Increased levels of analysis relating both to additional information provided by States in Additional Protocol declarations and information from other sources is an integral part of devising and applying State-Level Integrated Safeguards. Such analysis enables the development of relevant objectives to be addressed through safeguards activities and, in particular, complementary access.

A State-Level Integrated Safeguards Approach should include a plan for performing an adequate number of CAs, amongst other purposes, these can:

- contribute to providing assurance of the absence of undeclared nuclear materials and activities;
- confirm the decommissioned status of facilities;

- ensure that sensitive facilities are being operated in a manner consistent with that declared; and,
- assist in assessing the consistency of R&D activities with declared fuel cycle activities and objectives.

## 4. Conclusions

'Integrated safeguards' describes the combination of traditional safeguards activities with additional measures provided for under the additional protocol in a manner which optimises the efficiency and effectiveness of safeguards.

In order for the some of the traditional safeguards activities to be offset by complementary activities as occurs under integrated safeguards, it is first necessary for the Agency to have drawn the so-called 'broader conclusion' i.e. to have confirmed that in addition to the absence of diversion of declared material or misuse of declared facilities, no undeclared nuclear material or relevant facilities exist within a state. Once such a conclusion has been made certain diversion or proliferation pathways can be excluded from consideration enabling a relaxation in certain safeguards parameters.

In considering the approach to integrated safeguards in a state with developed nuclear fuel cycle activities it may be fruitful to consider a sector based division of facilities based on material flow patterns and similarities in material types at different facilities. Such an approach, when coupled with frequent advanced declarations by a state concerning the amount and type of nuclear material at facilities, may be used in combination with a randomisation approach to verification activities which enables the number of inspections to be reduced without a similar reduction in effectiveness.

Randomised verifications are effective both through their ability to address verification of chosen strata with a defined level of detection probability and also through deterrence in which aspect they have an impact on far more nuclear material than that actually verified.

Additionally, complementary access may be used as part of the approach, both as a wide spread deterrent to misuse of a large range of facilities and other locations and also as a means of following-up particular issues or inconsistencies.

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# Session 8

# CS-I

# Containment Verification Using 3D Laser Technology

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***Abstract:***

*A novel system for detecting tampering on nuclear containers, in particular on MOX fuel transportation flasks, is described. As a complementary technique to seals and surveillance, it can be employed to maintain continuity of knowledge between inventories and physical inspections. The proposed system uses laser range imaging in order to create 3D surface maps of the entire container surface, at a lower resolution, as well as of any weld seams at higher resolution. Reference surface maps are created and securely stored, for example upon manufacturing and/or sealing a flask. For subsequent verification, random parts of the welds and of the surface are scanned, and the scans are compared to the reference. Both for the weld and for the surface verification, a key requirement is that no high-precision mechanical alignment or calibration of the instrument be necessary. The user needs to manually set up the instrument in front of the container and to immediately start acquiring scans. For the case of the weld verification, this is solved by first correcting for alignment imperfections and then extracting certain "feature functions", such as the weld width, height or cross-section along the scan. These feature functions create a unique "fingerprint" of the weld. The fingerprint of a verification scan can be efficiently matched against a database of reference fingerprints, regardless where along the weld the verification scan has been acquired. For the case of the surface verification, we propose an approach for automatic registration of two scans acquired from different scanner viewpoints. The approach exploits the fact that the container surfaces are known to be cylindrical and does not require any user interaction such as marking corresponding point pairs. We present examples of both the weld and the surface verification and conclude that containment verification using 3D laser technology allows reliable detection of tampering.*

**Keywords:** containment verification; surface mapping; laser scanning

## 1. Introduction

Sealing and surveillance techniques are routinely used to ensure continuity of knowledge between physical inspections and inventory verifications [1]. There is, however, an increasing demand for containment verification techniques in order to detect any tampering with the nuclear container itself, circumventing the door or lid which can effectively be protected by tamper indication devices.

Three-dimensional laser range imaging has been proposed in the context of nuclear safeguards, e.g. for unique identification of objects [2]. In this paper, we propose an approach to containment verification using 3D laser range imaging in order to record the precise structure of the container surface as a reference. Upon subsequent verification, a new scan of the surface is acquired and matched against the reference model. Any changes to the shape can be detected at sub-millimetre accuracy. We argue that this makes an undeclared penetration of a nuclear containment at least very difficult since any

attempt to open the container and to restore its structure is bound to leave slight changes to the three-dimensional shape.

Particularly vulnerable parts of a container are any seams, such as weld seams on metal containers. It is conceivable that a metal container is cut open along a weld seam and subsequently rewelded, leaving the surface of the container perfectly intact. However, such an attack scenario would change the reflow pattern of the weld which can also be detected by means of laser scanning with appropriate resolution and accuracy.

The principle of laser triangulation is illustrated in Figure 1: A sheet of light is projected onto the object using a laser diode and a cylindrical lens. Where this sheet of light intersects with the object surface, it creates a laser line which is viewed by a digital camera from a different angle. The shape of the laser line "seen" by the camera depends on the shape of the object. Assuming the system is properly calibrated, each point of the laser line recorded by the camera yields the coordinates of one point in the laser plane. Therefore, each camera image yields a profile which is the intersection between the laser plane and the object surface. By moving the object or the scanner in a controlled manner, a sequence of profiles can be acquired, yielding a dense cloud of 3D points on the object surface.

Note that the requirements in terms of resolution and accuracy on the one hand and acquisition speed and memory consumption on the other hand differ strongly between the weld seams and the enclosure as such. We therefore pursue an approach using two different sub-systems for the welds and for the enclosure. While the measurement principle is the same for the two sub-systems, the instrumentation as well as the software for the three-dimensional reconstruction, registration and matching are specific to the two different applications. Section 2 describes our solution for weld identification and authentication while the system for the integrity verification of the enclosure is presented in Section 3. Section 4 contains concluding remarks and an outlook on our future work in this field.

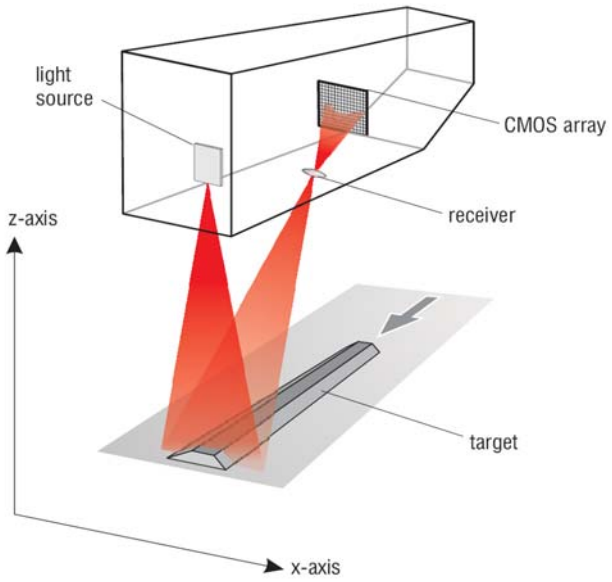


**Figure 1:** Measurement principle of laser triangulation (figure courtesy of Micro-Epsilon).

## 2. Tamper detection for weld seams

The welds of a nuclear container are particularly obvious attack points since cutting and rewelding a weld seam would not leave any traces which could be detected by visual inspection. Verifying the identity and integrity of welds using Eddy current measurements has been proposed by Tolk and Stoker [3]. The use of laser scanning has been discussed in [4] and [5]. However, the approach of [4, 5] is based on scanning a very small and known piece of a weld. Between reference acquisition and verification, the exact same piece of the weld needs to be scanned. We present a more general approach here based on feature extraction and matching which allows the use of a randomly selected piece of a weld for verification. The approach is described in detail in [6] and has been tested in laboratory experiments with Plutonium canisters as well as with large steel barrels.

## 2.1. Instrumentation

The setup for our laboratory experiments is shown in Figure 2: The laser scanner remains in a fixed position while the Plutonium canister is rotated in front of the scanner. For the large steel barrels, the scanner was moved along the welds on a rotating arm. We used a commercial off-the-shelf laser scanner whose technical characteristics are listed in Table 1. Note that it uses an eye-safe class 2M laser source. The scanning speed obviously depends on the desired profile density, or spacing between adjacent profiles. A natural lower limit for the profile-to-profile spacing is the width of the projected laser line (approx. 70 µm). Scanning at a higher profile density than given by the laser line width does not yield much additional information. At this maximum profile density, the instrument can scan at a speed of 8.75 mm/s. Our tests, however, have shown that the profile spacing can be increased by up to one order of magnitude without strongly compromising the system performance, i.e. a scanning speed of up to 87.5 mm/s is feasible.

Figure 3 shows an example of a range image acquired with this setup. Each column of this range image corresponds to a single profile captured by the scanner. Note that Figure 3 is a false-colour image where the colour of each pixel represents the “range”, i.e. the distance in z-direction of the respective surface point from the origin of the scanner coordinate frame. In this example, the image is scaled to a range of 1.4 mm, i.e. a white / bright yellow pixel is 1.4 mm farther away from the scanner than a black/dark red pixel. It can be seen that the 3D structure of the reflow pattern of the weld is accurately captured.

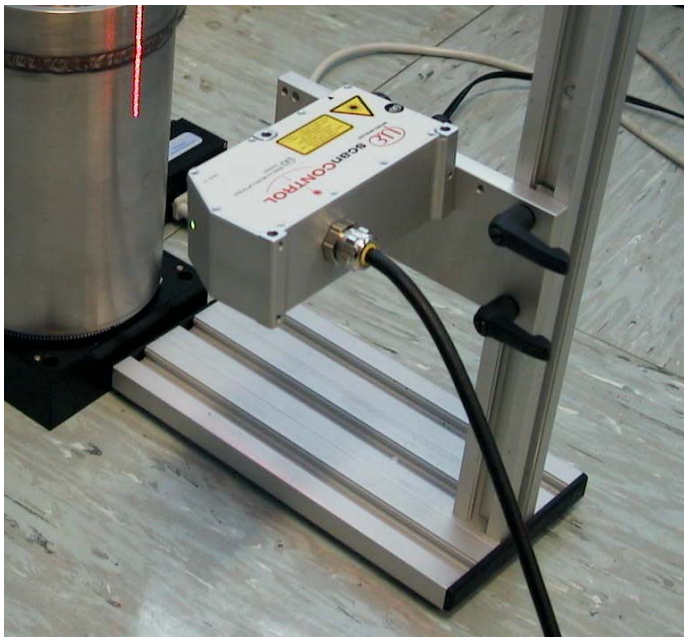

**Figure 2:** Instrument for scanning weld seams.

| **Laser scanner** | **Micro-Epsilon LLT 2800-100** | |
|---|---|---|
| | **Near** | **Far** |
| Measurement range, z-axis | 115 mm | 246 mm |
| Measurement range, x-axis (“length” of profile) | 50 mm | 140 mm |
| Resolution, x-axis (point-to-point distance in a profile) | 49 µm | 137 µm |
| Resolution, z-axis | 40 µm typical | |
| Profile frequency | 125 profiles / s | |
| | **Slow** | **Fast** |
| Profile-to-profile distance | 70 µm | 700 µm |
| Scanning speed | 8.75 mm / s | 87.5 mm / s |
| Laser wavelength | 655 nm | |
| Laser class | 2M | |

**Table 1:** Key technical features of the weld scanning system of Figure 2.

## 2.2. Image processing algorithms

Once a reference scan and a verification scan of a weld have been acquired, they need to be compared to each other. A direct point-to-point comparison, however, would only be possible in a situation of perfect mechanical alignment between the two scans. In practice, however, the system should be set up manually without the need for any calibration or high-precision mechanical adjustments.

Therefore, we must allow for a number of mechanical imperfections: The distance of the scanner from the surface, the point along the weld where the scan was started, the position and orientation of the scanner with respect to the surface may all vary. We therefore pursue an approach which is described in more detail in [6]: First, corrections are made for mechanical imperfections. For example, the distance between the scanner and the surface as well as the angle between the scanning direction and the surface normal are corrected for by normalizing both the reference scan and the verification scan. Figure 4 shows the range image of Figure 3 after pre-processing. The surface to both sides of the weld is now normalized to zero, and any range gradients across this surface have been removed.

Next, for each profile along the weld a number of features are extracted. The course of each feature along the weld yields a feature function. By extracting a number of features per profile, several feature functions are generated which create a unique “fingerprint” of the pattern of the weld. We have found a very simple set of feature functions, illustrated in Figure 5, to yield good results: It consists of the width, height, and cross-section area of the weld per profile. Figure 6 shows the three corresponding feature functions along a piece of a weld. This set of functions serves as a unique “fingerprint” which is used to uniquely identify and authenticate the weld. Not surprisingly, the height and area features are closely correlated while the width yields a rather uncorrelated, additional feature function.

In order to compare two fingerprints to each other, it still needs to be established at which position along the weld the two scans have been started. In other words, there may still be an unknown shift between the two fingerprints. In [6], we introduce an error measure between two fingerprints which is computed for each possible shift. Figure 7 shows these error functions for the three extracted features for the example from Figure 6. It can be seen that the error functions of all three features exhibit a very pronounced minimum at the same shift of approximately 40 mm. Note that the vertical axis is scaled logarithmically and the minima are almost two orders of magnitude below the “noise floor”. Ideally, the error function would be zero at the matching position – the remaining error stems from the measurement uncertainty of the scanner and any error residuals from the preprocessing and feature extraction steps. The errors at the non-matching positions result from the variation of the reflow pattern of the weld itself.

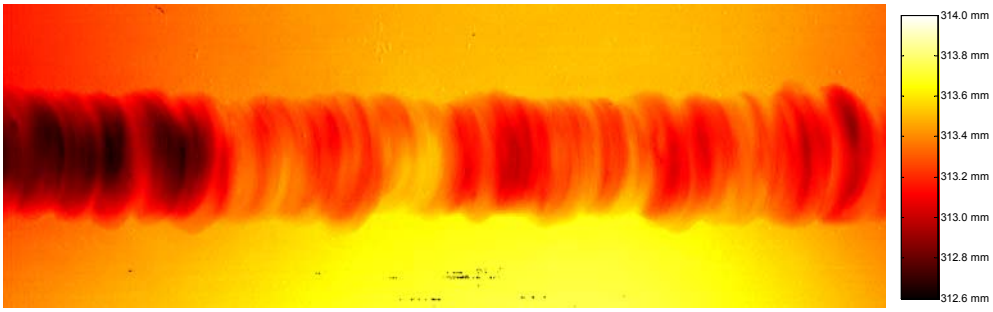


**Figure 3:** Raw range image of a weld.

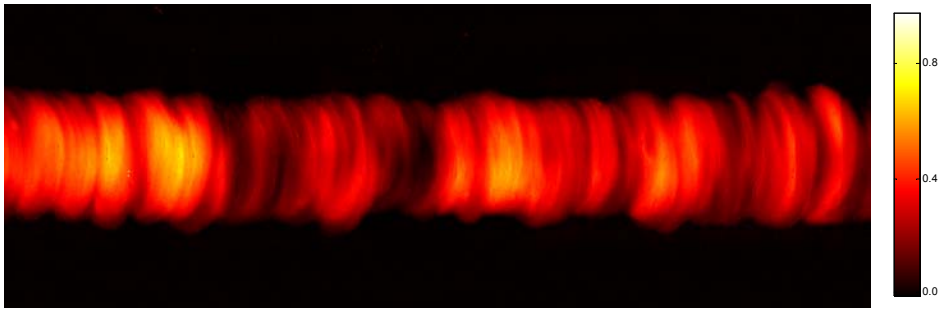


**Figure 4:** Range image from Figure 3 after pre-processing.

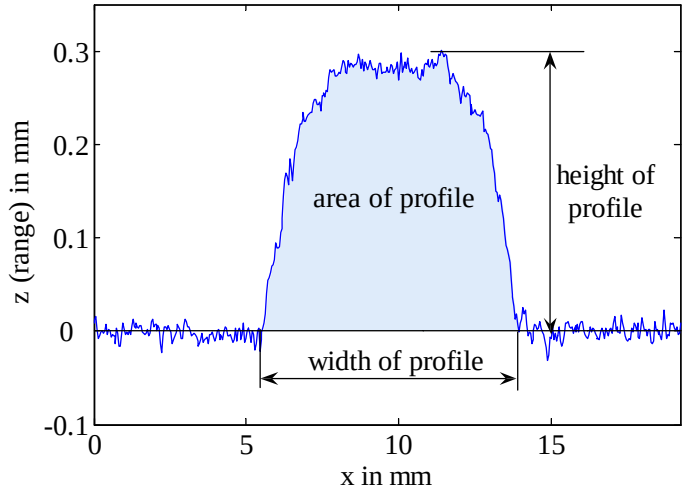


**Figure 5:** A single profile of the range image from Figure 4. It is indicated in the figure how the width, height and cross-section area features are extracted from each profile.

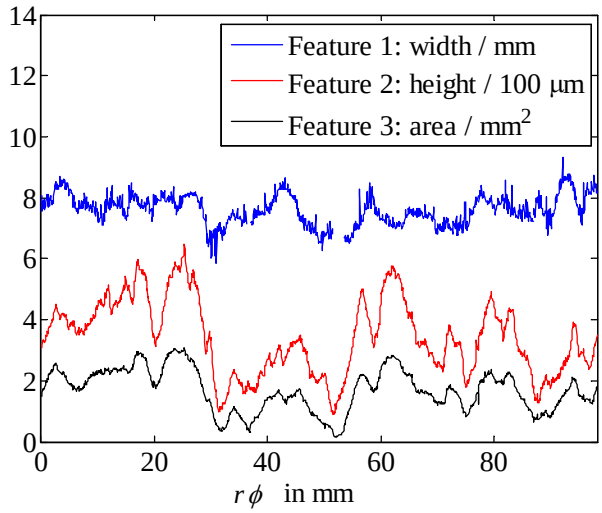


**Figure 6:** Width, height and area feature function along a piece of a weld. The combination of the three feature functions serves as a unique "fingerprint" of the weld.

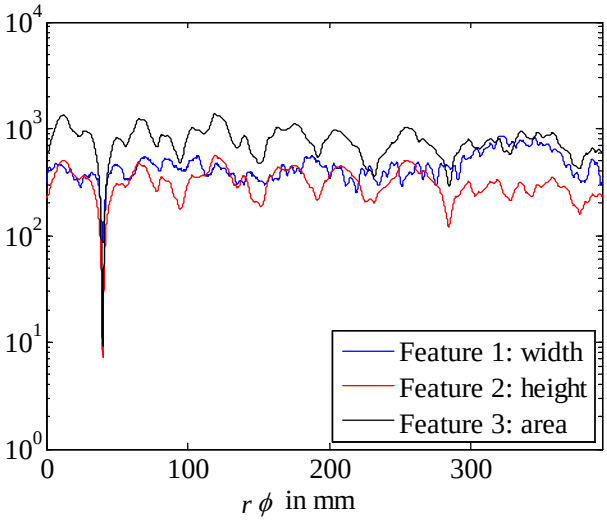


**Figure 7:** Error functions for width, height and area feature functions between the fingerprint of Figure 6 and a corresponding reference fingerprint. It can be seen that all three feature functions match at a shift of $r\Phi$ = 40 mm.

### 2.3. Results

Results have been presented in detail in [6]. We have performed laboratory tests with six Plutonium canisters. When combining the error functions of the three feature functions, we typically obtained separations of three orders of magnitude between a matching and a non-matching weld; the worst-case scenario was still almost two orders of magnitude.

We have further shown that the minimum length which needs to be scanned during verification, was 30 mm. Below this value, the risk of a false match (i.e. the system accepts two welds as matching even though they are not identical) increases strongly. It should be noted, however, that this minimum length is likely to vary with different types of welds (width, material, welding technology). Further, our results indicate that – for the welds used in our laboratory experiments – the profile-to-profile distance can be increased up to about 700 µm without a substantial loss in performance. This allows increasing the scanning speed to about 87.5 mm/s, i.e. a verification scan can be completed in around 340 ms.

In summary, the system has been shown to work reliably and efficiently for Plutonium canisters, rotating on a turntable. Results are similar for other types of welds and containers, even though other mechanical solutions are needed in this case.

## 3. Tamper detection for container surfaces

### 3.1. Instrumentation

A different instrument, shown in Figure 8, is used to scan the enclosure surface of the nuclear containers. Obviously, the field-of-view and scanning speed of the weld scanning instrument are not sufficient to allow an efficient scanning of larger surfaces. In the system in Figure 8, the laser line scanner is mounted on a rotation table while the container is not moved during the scan. The technical features of the scanner are listed in Table 2. In this case, we found it necessary to use a class 3B laser. Experiments with a class 2M laser yielded poor results for containers which were either painted with dark colours or completely unpainted.[1] Appropriate eye safety measures need to be implemented. The instrument scans a field-of-view of approximately 800 x 800 mm in 4.5 s.

### 3.2. Image processing algorithms

The image processing algorithms developed for this type of instrument have been presented in detail in [7]. Two important processing steps are necessary before a reference scan and a verification scan can actually be compared to each other:

**3D-reconstruction and calibration:** Each profile acquired by the rotating laser line scanner needs to be correctly remapped into a common 3D coordinate frame. This can be done if the position of the rotation table at the time of the profile acquisition is known, as well as the exact position and orientation of the rotation axis with respect to the internal coordinate system of the scanner. Since it is not realistic to mechanically measure the position and orientation of the rotation axis on the one hand and of the scanner coordinate system on the other hand at the sum-millimetre accuracy required, we have proposed an auto-calibration scheme in [7]. It is based on acquiring several scans from different viewpoints of a planar calibration target which is known to have a very high planarity (e.g. an optical breadboard or a coated glass pane). The sought calibration parameters are then numerically computed in such a way that the planarity of the 3D-reconstructions of the calibration target is maximized. We have shown that with this method it is possible to accurately calibrate the system and to obtain true 3D surface maps.

[1] Note that the laser scanning technology as shown in Figure 1 relies on the diffuse ("Lambertian") scattering of the laser light on the surface; similar to the scatter that occurs on a sheet of paper which is approximately direction-independent. On surfaces with dominant specular reflection, such as unpainted, polished metal, the laser line will appear extremely dark to the camera, except if the surface inclination happens to be at the angle of specular reflection between the laser line and the camera. Therefore, shiny surfaces are even more difficult for laser triangulation scanners than dark surfaces.

**Figure 8:** Instrument for scanning the container surfaces.

| **Laser scanner** | **Sick IVP Ruler E600 B** | |
|---|---|---|
| | **Near** | **Far** |
| Measurement range, z-axis | 415 mm | 865 mm |
| Measurement range, x-axis ("length" of profile) | 450 mm | 820 mm |
| Resolution, x-axis (point-to-point distance in a profile) | 293 µm | 534 µm |
| Resolution, z-axis | 150 µm typical | |
| Profile frequency | 450 profiles / s | |
| Profile-to-profile distance | 400 µm | |
| Scanning speed | 180 mm / s | |
| Laser diode | 660 nm | |
| Laser class | 3B | |

**Table 2:** Key technical feature of the surface scanning system of Figure 8.

**Registration:** In general, a given reference and verification scan will have been acquired from different viewpoints and scanner orientations. Therefore, even after 3D-reconstruction, the two scans cannot be directly compared to each other. First, we need to find the correct "rigid transformation", i.e. combination of a shift and rotation in 3D such as to make the two scans match best. Standard solutions to this problem exist, but they cannot be directly applied here: The objects are cylindrical and hence have certain symmetry properties. Typically, a standard registration algorithm will succeed in aligning the two surfaces, but will fail to find the correct shift along the cylinder axis as well as rotation about the cylinder axis between the two scans. In [7] we have proposed an algorithm which is specific to cylindrical surface patches and which finds the correct solution for these two degrees of freedom. It first uses a robust fitting algorithm to fit a parametric cylinder model to the scan (Figure 9). Once the cylinder model parameters, i.e. the cylinder axis and radius, are known, the surface can be unfolded into a 2.5D-representation as the one in Figure 10. In this range representation, the value of each pixel specifies the distance of the true surface from the ideal cylinder. The vertical axis is the cylinder axis ($z'$-axis) while the horizontal axis corresponds to the angle in the cylindrical coordinate system ($\varphi$-axis). In this representation, the correct shift along the cylinder axis and rotation about the cylinder axis become vertical and horizontal image shifts, respectively. The correct shift can be automatically

found by identifying “feature windows” with a characteristic surface structure and matching them between the reference image and the verification image by means of correlation techniques [7]. Suitable feature windows – the red boxes in Figure 10 – are surface regions with a rather unique and characteristic structure. In Figure 10, it can be seen that feature windows have only been selected along the two welds. It is also conceivable to use the unfolded intensity images instead of – or in addition to – the unfolded range images for the recovery of the two unknown parameters. Once the correct shift between the two images has been found, one can apply the corresponding transformation in 3D, so that then the reference scan and verification scan can be compared to each other.

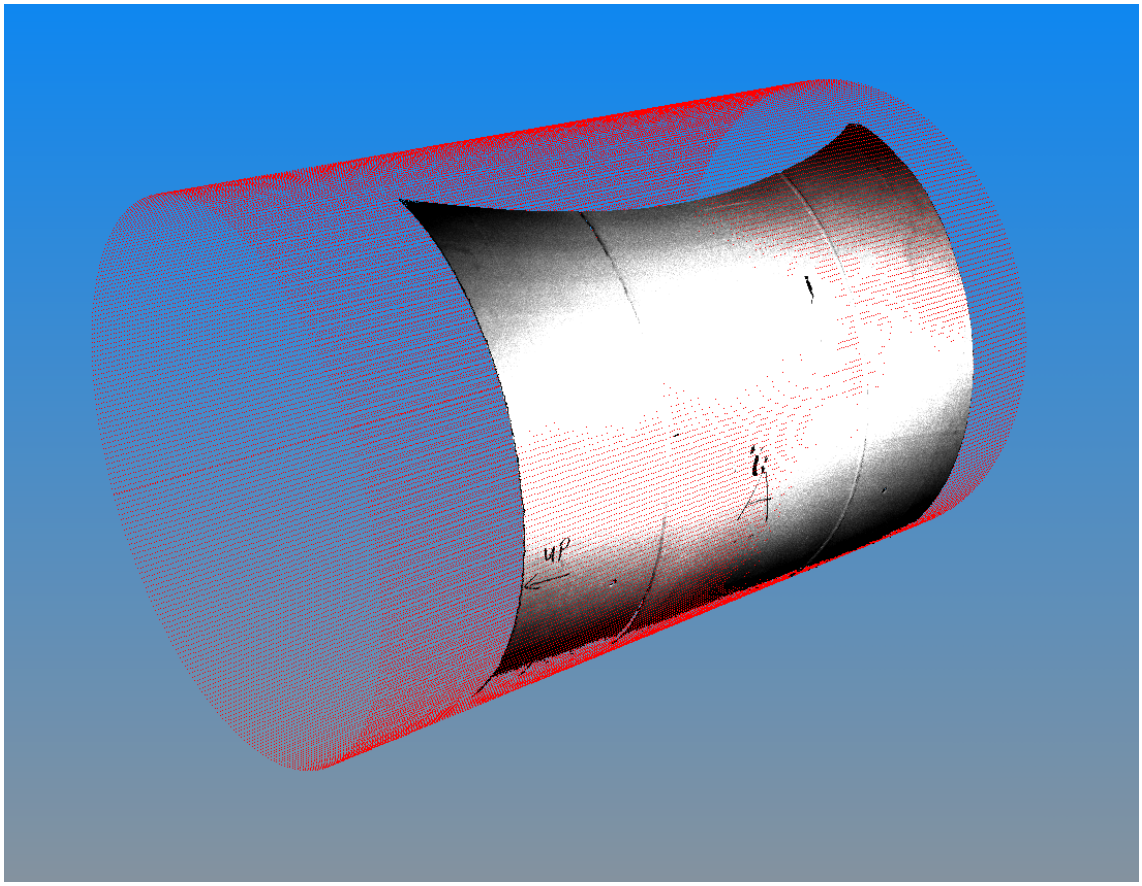

**Figure 9:** 3D reconstruction of a patch of a cylindrical container (gray) and cylinder fitted to this surface (red).

### 3.3. Results

Two example results are shown in Figures 12 and 13. The colour of each point indicates the distance of the respective point in the verification scan from the surface of the reference model. Figure 11 shows the case of a “matching” inspection, i.e. the reference and verification surfaces were identical. The blue and turquoise colours show low distances – after successful registration – between the reference and the verification surface. The system noise increases towards the edges of the field-of-view as well as in areas of strong surface structure, e.g. welds. It should be noted that the two scans were acquired from completely different scanner viewpoints.

Figure 12 shows an example where a business card was glued to the surface in between the acquisition of the two scans. The business card can be clearly seen in the figure as a change in the surface structure well above the noise level.

We have further experimented with steel as well as concrete containers, attempting to cut them open and to remodel the surface as accurately as possible. In both cases, we ended up leaving very strong changes to the surface shape, well above 1 mm, i.e. the surface inspections would exhibit extended bright-red areas. It should be noted that in both cases, the changes are not perceptible by the human eye. A human observer cannot tell the difference between an intact surface and one that was tampered with.

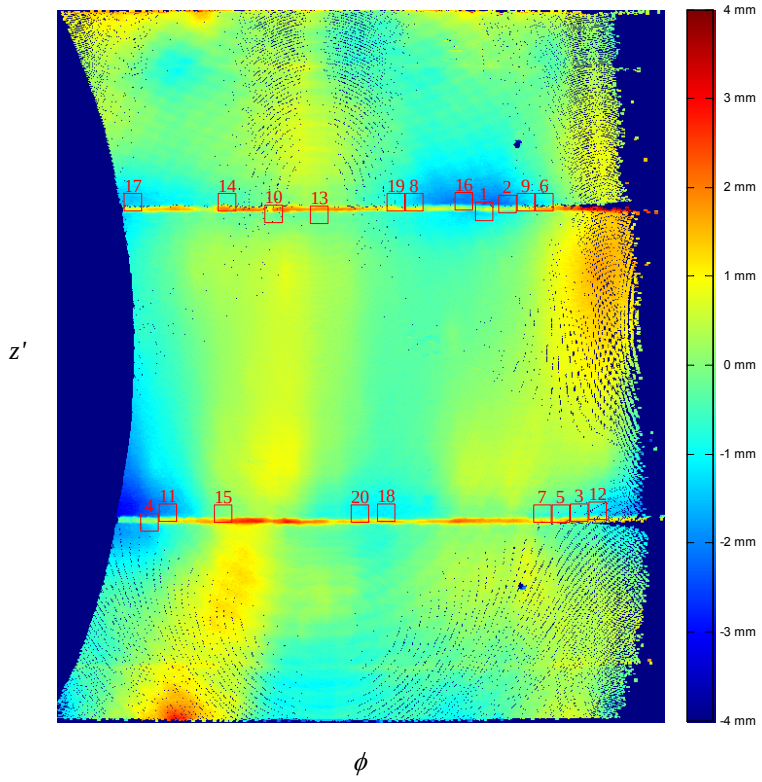


**Figure 10:** Range image of the unfolded cylinder surface. The colours indicate the distance of the surface from the ideal cylinder. Two horizontal welds can be clearly seen in the image. The red boxes indicate the feature windows which have been automatically chosen for recovery of the shift along and rotation about the cylinder axis.

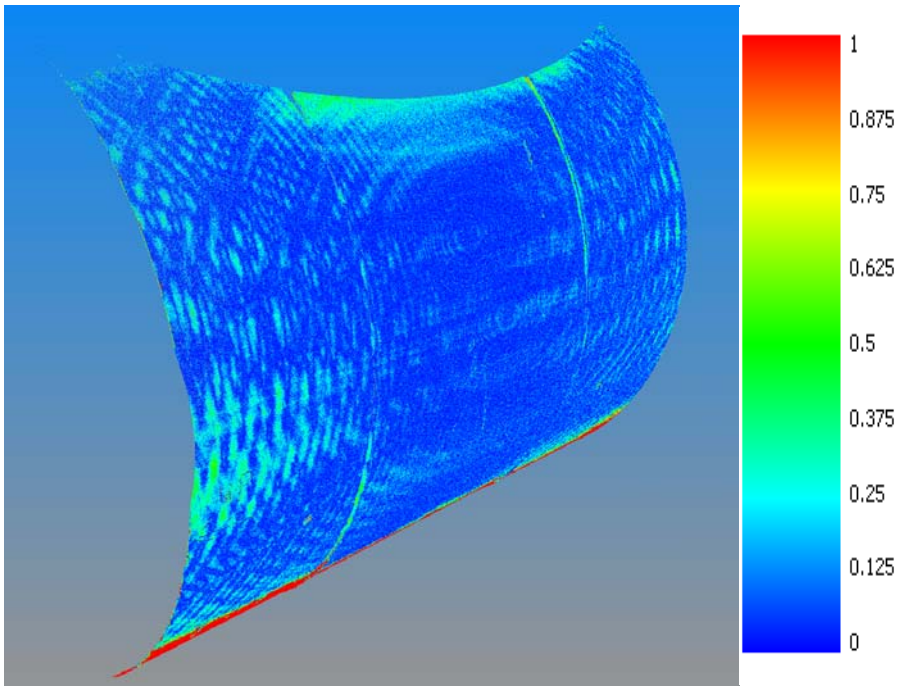


**Figure 11:** Inspection of a matching surface. Note that the reference scan and acquisition scan were acquired from different scanner viewpoints. It can be seen from the blue and turquoise colours that the two surfaces are identical to the accuracy of the scanning instrument.

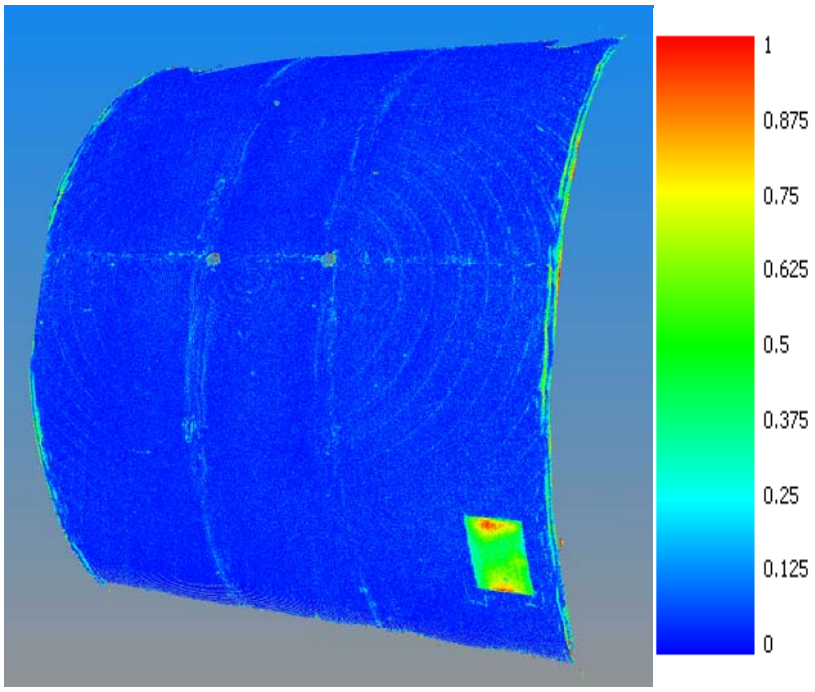


**Figure 12:** Verification of a non-matching surface. A business card was glued to the surface for the verification scan which had not been present in the reference scan. The business card can be clearly identified as an area where the distance between the two surfaces is approximately 0.5 mm.

## 3. Conclusion and Future Work

Laser scanning technology can detect changes to the three-dimensional shape of a surface at sub-millimetre accuracy. While there is no such thing as perfect security, the proposed approach at least makes it very difficult to intrude a container without leaving any detectable traces. In combination with sealing, surveillance and potentially complementary techniques for containment verification, it can provide a high level of protection against tampering with nuclear containers. We have shown both for the case of the container enclosure and specifically for the case of weld seams that tampering can be effectively detected with the proposed approach.

Our future work is focused on finding practically deployable mechanical solutions, both for the enclosure and for the weld seam verification. For the enclosure verification, we have presented an easy-to-use and portable system which can be used to acquire verification scans. However, other mechanical solutions will be necessary to allow the acquisition of a coherent reference model of an entire cylinder in reasonable time and with as little user interaction as possible.

For the weld scanning, it is necessary to guide the scanner along the weld at high accuracy, or at least to precisely record – for each profile – which scanner position and orientation is has been acquired from. While we have found elegant solutions for small objects such as Plutonium canisters, other approaches have to be pursued for larger containers. Certainly, this end can be achieved using a robotic arm moving the scanner. Our current work, however, focuses on finding mechanical solutions which are more cost-efficient than a full-scale robotic solution, yet more versatile than a simple turntable. The ultimate long-term goal would be a handheld scanner which can be manually moved along the weld and which is able to accurately track its own position and orientation.

## Acknowledgment

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# 3D Reconstruction in Nuclear Security

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**Abstract**

Accurate modelling is gaining increasing importance in security applications. Indeed, realistic and dimensionally accurate models of critical areas can be used for prevention and simulation exercises as well as for planning emergency responses once an attack is perpetrated. CEA-DAM and EC-JRC engaged in a joint exercise involving the simulation of a terrorist attack in an urban area with possible release of radiological substances. JRC was responsible for creating a dimensionally accurate (centimetre accuracy) 3D model from the urban area "as-is" before and after the attack. Further tests involved the automatic 3D detection of changes in both indoors and outdoors environments. The paper describes the principles and technologies behind the generation of photo-realistic and accurate 3D models of wide areas "as-is", and will discuss the use of those technologies for nuclear security applications.

Keywords: Nuclear Security, 3D Reconstruction, Data Integration, Change Detection

## 1 - INTRODUCTION

The French Commissariat à l'Energie Atomique, Direction d'Applications Militaires – CEA-DAM – and the European Commission's Joint Research Centre, Institute for the Protection and Security of the Citizen – JRC-IPSC – engaged into informal discussions on technologies relevant for security research. From these discussions, JRC's three dimensional laser-based technologies were identified as of interest to future security applications.

CEA-DAM invited JRC-IPSC to participate in a joint exercise involving the simulation of a terrorist attack in an urban area with possible release of radiological substances. JRC was responsible for creating a 3D model from the urban area "as-is" before and after the attack. To this effect a JRC team travelled to the exercise site and made a photo-realistic and dimensionally accurate 3D reconstructed model using laser range and photography scanning. 3D Data acquired from different viewpoints were registered and integrated with the photographic records into a single geometric model with centimetre accuracy. Following the "attack", specific spots of the urban area were again scanned and new models created. Several practical applications of these 3D models are currently being considered:

a) Documentation of the area under attack - before and after;
b) Detection of changes;
c) Geometric input to atmospheric dispersion models;
d) Calibration of atmospheric dispersion models in what concerns the propagation and diffusion of smoke plumes and clouds;
e) Generic platform for the intuitive presentation and integration of data from multiple sources and time-frames (e.g., ground penetrating radar, satellite and airborne images and 3D scans, radiation and thermal images, CAD data, etc.)

This paper describes the investigations done, including the results obtained in the

experimental campaign. It also looks forward in terms of future security applications.

## 2 - 3D RECONSTRUCTION

Three dimensional reconstruction – 3D Reconstruction – refers to the techniques that allow the creation of dimensionally accurate geometric models of real environments both indoors and outdoors, including small objects. For several years JRC has been developing techniques to this effect [1, 2]. Most of the techniques are based on the use of laser based instrumentation to generate 3D cloud of points of the object or environment to reconstruct. These clouds of points contain the 3D information as perceived from the sensor. In general, to reconstruct an environment more than one single-view capture point is necessary as some objects hide – i.e., occlude – others.


**Figure 1: (a) Laser scanner on tripod with dolly for indoor operation; (b) Laser scanner mounted on vehicle for outdoor operation.**

Figure 1 shows two commercial, off-the-shelf, laser range scanners that are used for 3D data collection. Whereas the laser on the left (Fig. 1.a) is mainly used for indoor applications the laser on the right (Fig. 1.b) is generally used for outdoor applications. This division is mainly the result of considerations on accuracy, spatial resolution and maximum measured range. Table 1, illustrates the main features for both instruments. The reconstruction of small objects is mainly obtained using triangulation based laser scanners though the details are beyond the scope of this paper.

| | **Z+F Imager 5** | **Riegl Z 420i** |
|---|---|---|
| **Range** | 53.5 m | 2 - 250m ($\rho$ > 10%) or 800m ($\rho$ > 80%) |
| **Maximum Acquisition Rate** | 500,000 points per second | 12,000 points per second |
| **Accuracy** | 4mm | 10mm |
| **Field of View** | 310° (vert.) by 360° (horiz.) | 80° (vert.) by 360° (horiz.) |
| **Spatial Resolution - Vertical** | 55.5 samples/° (17,205 samples) | 125 samples/° (10,000 samples) |
| **Spatial Resolution - Horizontal** | 55.5 samples/° (20,000 samples) | 100 samples/° (36,000 samples) |
| **Output** | range and reflectance | range and reflectance |
| **Eye Safety** | Class 3R | Class 1 |

**Table 1: Main features of the two laser scanners used for 3D modelling**

Once an environment is scanned there is the need to:

a) Get a single coordinate frame to the data from single-view scans – registration:
b) Combine the data from multiple scans and, eventually, discard redundant data
c) Integrate other types of data into the 3D reconstructed model. Important pieces of data are photos from the environment to create visually realistic models

All the above operations are done using JRC's *3D Reconstructor*® software package. Figure 2 describes the architecture/paradigm behind the complete and realistic 3D reconstruction of environments. The *3D Reconstructor* package implements the different processing blocks.

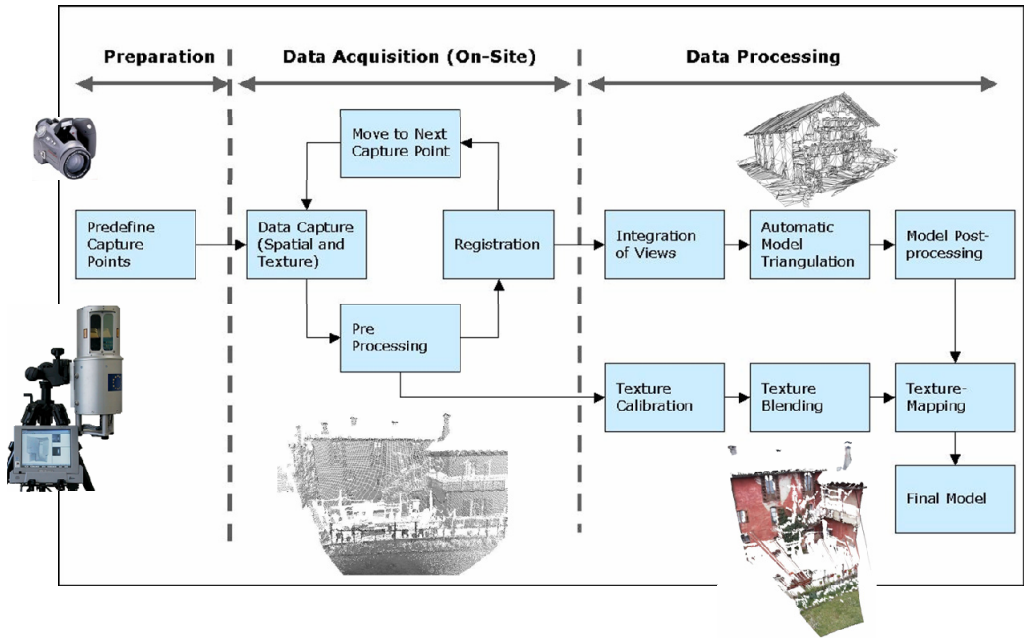


**Figure 2: System's Architecture of JRC's *3D Reconstructor* ®**

To illustrate a practical use for *3D Reconstructor* technologies one should mention its routine use by the International Atomic Energy Agency (IAEA) for design information verification (DIV) purposes. DIV aims at accurately mapping interiors of complex nuclear facilities under safeguards. DIV activities are performed during the complete life cycle of a nuclear facility to confirm that it operates as declared and is not misused for undeclared nuclear activities [3].

## 3 - JOINT NUCLEAR SECURITY EXERCISE

CEA's emergency response project organized an internal – to the CEA – security exercise involving the participation of radiological and nuclear experts from several sites. The exercise was triggered by an 'alarm' indicating that a "terrorist accident with release of radioactive/radiological substances had taken place at the CEA-CESTA research centre."

The "attack" consisted of having two cars burn/explode in an urban area, with simulated release of several samples of radiological and/or radioactive material. Different teams from different CEA sites immediately travelled to CESTA. The teams brought with them a van with NDA and radiological detection equipment, as well as a car equipped for detecting radioactive substances in urban environments. All measurements are geographically referenced using on-board GPS.

**JRC's participation:** During the exercise the JRC team created 3D models from the urban environment before and after the simulated terrorist attacks. This included scans from:

a) an indoor environment – warehouse – to document the premises and detect changes;
b) the urban environment – outdoors – for 3D site reconstruction;

c) the burning vehicles – outdoors – to investigate the use of 3D information to model the smoke plumes and clouds.

The following sections will detail the three independent exercises

**Figure 3: View of the 3D reconstructed model of a warehouse. Ten laser scans were used to build this model, taking approximately 60 minutes. All points were measured with an accuracy of 4mm. Light blue indicates areas for which no 3D information is available (i.e., have not been scanned).**

## 4 - INDOOR ENVIRONMENT

The objectives of this exercise were twofold:

i) to create a dimensionally accurate and visually realistic model of an indoor environment that could be used for documentation purposes;

ii) to detect minor changes occurring in this quasi-static environment

The warehouse was scanned from ten different positions with the Z+F laser scanner. For each scan a set of seven photographs were taken (total: 70 photos). With both the range and photo images a millimetre accuracy model of the environment was created. Figure 3 illustrates the 3D model of the warehouse.


(a) (b) (c)

**Figure 4: Detection of changes inside the warehouse. Changes are indicated in red/orange: (a) changes in the shelves area; (b) changes in warehouse's main hall – three objects were detected: a mobile phone on the shelf (bottom left), the crane cable (top right) and the forklift (centre-right); (c) false colour code – red corresponds to distance changes larger than 10cm; change in the object in (a) is about 6.5cm.**

Two more scans were done to detect minor changes. It should be noted that none of the changes introduced in the environment were known to the team. All changes were detected. Figure 4 shows some of the changes detected in red/orange. These include (a) changing the position of a mobile phone on a shelf; (b) changing the positions of a few boxes on the shelves; (c) rotating a pot with a plant; (d) changing the position of the crane; (e) changing the position of the forklift vehicle.

The total time for acquiring the 3D and visual data for ten modelling scans and two verification ones, was about 90 minutes. Processing time amounted to about 3 hours.

**Figure 5: Partial 3D Overview of the CESTA site. Eight laser scans were used to create this model. Each model point has an accuracy of about 2cm. Processing time amounted to about 3 hours.**

## 5 - OUTDOOR URBAN ENVIRONMENT

In this case, and considering the dimensions and complexity of the wide area environment the work was divided in two parts:

**Part I: Build a general overview model of the site** – this objective involved the creation of a 3D model of the site layout as "seen" from the existing water tower which was then updated by local scans acquired on the ground. Figure 5 shows three snapshots of the site model. The model integrates data acquired from the water tower and 7 scans acquired at ground level.

**Part 2: Document the "terrorist" attack scene –** this objective was met by scanning from several viewpoints – to resolve for possible spatial occlusions – the street where the "terrorist" attack took place. This was done before and after the attack. Figure 6 illustrates these scans.

**Figure 6: View of the scene before the 'attack'.**

Given that in a few cases the range of distances was inferior to 50meters, both the indoor and outdoor laser scanners (see Figure 1) were used. JRC's *3D Reconstructor* software can easily merge and integrate data from scans with different spatial resolution.

## 6 - PLUME DETECTION AND MODELLING

Any explosion attack is normally associated with a fire and corresponding smoke. Depending on the explosion, dangerous chemicals may be freed into the open atmosphere. It is thus of interest to predict how such dangerous cloud will disperse in the air. This knowledge is particularly relevant in urban areas to:

(a) help deciding where to concentrate rescue effort (resource management)
(b) time allowing, alert the population

Cloud dispersion is a complex modelling exercise, especially at short ranges, as it depends on many factors such as, wind direction and speed, air temperature and humidity, gases to be dispersed, and on the specific 3D topology of the environment both where the explosion took place and of its vicinity. During the preparation of this joint exercise it was found useful to make a few experiments in order to provide quality data to assist the atmospheric dispersion modelling.

**3D Modelling of the Environment:** the laser scanning technique and 3D Reconstuctor software provide easily a complete modelling of the environment "as-is" with an accuracy of about 2cm. Such 3D model can be used as input in any short-range atmospheric dispersion model.

**3D Plume Detection and Modelling:** Many 3D scans of the plumes were made in order to find out whether the laser scanning equipment could (i) detect the plume itself and (ii) trace its dispersion in terms of both height and orientation.


**Figure 7: Two snapshots of the 3D model of the smoke plumes emanating from the burning car on the left. The laser scanner is located at the right. The dashed lines indicate the scanning field of view (FoV). Only the smoke plumes inside the FoV were captured.**

The two laser scanners – indoors and outdoors – were used in the experiments. It was soon found out the indoors scanner could not always detect the plumes, probably due to the small diameter of the laser beam. Encouraging results were obtained with the outdoor laser.

It was possible to detect the plume, even when it was very dark (i.e., corresponding to poor laser reflectance). It should be noted that with the use of a single laser scanner it is possible to detect and map only one of the sides of the envelope of the smoke plume. In other words it is not possible to measure neither the plume diameter (i.e., thickness) nor its density. Further, changing wind speeds and the chaotic behaviour of the explosion fire makes plume dispersion a highly dynamic four dimensional phenomenon, i.e., the shape and orientation of a plume change in time. Figure 7 illustrates the capability for the 3D detection of the smoke plumes.

## 7 - FUTURE WORK:

**Combining 3D models with radiation maps –** It is possible to create dimensionally accurate and photo-realistic 3D reconstructed models from real environments both indoors and outdoors. This representation can be used in many ways, including documentation, detection of changes and for training purposes.

It is possible to use the 3D models as a human presentation and interfacing tool by integrating the 3D model with real measurements such as radiation or thermal maps. Figure 9 illustrates the results of experiments made in collaboration with US DoE's Oak Ridge and Lawrence Livermore National Laboratories [4]. In this particular case the 3D model of a room is integrated with both the photographic map and a gamma radiation map. It is possible to see the gamma measurements produced by a radioactive source inside one of the pipes hanging from the ceiling. Such combined representation eases the interpretation of the scene as it allows the easy location of relevant sources, including the accurate and interactive measurements of distances or volumes.

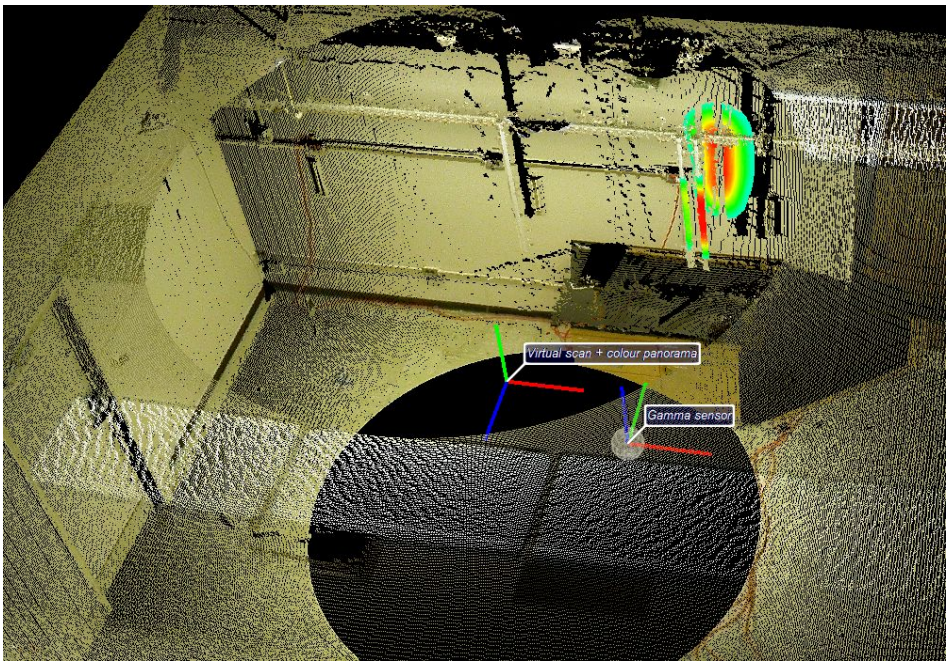


**Figure 9: Snapshot of the 3D and gamma mapping integrated model. The laboratory room has two pipes hanging from the ceiling. One of the pipes contains a radiation source inside. The positions of the gamma-ray imager and 3D scanner are represented.**

The same principle can be used outdoors integrating 3D and radiation mapping. Data can be acquired both at the ground level (e.g., using the vehicle of Figure 1.b) and airborne (e.g., using UAVs – unmanned aerial vehicles). The enhanced models serve as base lines (i.e., references) for subsequent verification scans to detect structural or radiation changes which might indicate situations and locate areas for detailed investigations.

**Combining 3D models with Ground Penetrating Radar –** GPR is gaining an increasing role as a security sensor. Indeed, its potential to detect cavities, hidden rooms or underground cells in buildings or tunnels in public areas is highly relevant to find hideouts or places where suspicious activities may be planned. As with many other sophisticated sensors, the output of GPR needs to be interpreted by expert staff. Both GPR and 3D laser scanner compute distances from the time analysis of an echo signal and their output corresponds to two datasets of 3D points. This parallel suggests that the integration of 3D laser range data with GPR could be beneficial for the global overview and correct interpretation of complex scenes.

**Combining 3D models with Airborne Imagery –** Most of the 3D models were based on data acquired at the ground level. This data provides good detail of buildings but fail to provide any detail about the roofs or building tops. Airborne data – either range or photos – do complement the data acquired at ground, i.e., at the street level and lead to the possibility to have complete 3D urban models. Different scans could then detect 3D changes occurring on building roofs or terraces.

## 8 - CONCLUSIONS

The following capabilities for JRC's 3D modelling system – indoors and outdoors – were successfully tested during the experiments. It was possible to prove the system's:

(a) **Portability**: the system is highly portable. It can be easily packed and transported in either a car or a plane
(b) **Fast deployment**: once an emergency occurs the system can be installed and made operational in a few minutes
(c) **Accuracy**: the system can model a real environment – both indoors and/or outdoors – and provide a dimensionally accurate representation (typical accuracies are 3mm indoors and 2cm outdoors)
(d) **Speed**: the system only takes a few minutes to scan an environment and produce a good quality documentation
(e) **Ease of Processing**: all data processing can be done in a normal laptop computer, i.e., there is no need for sophisticated and expensive hardware
(f) **Interoperability**: JRC's *Reconstructor*® software has many import and export filters enabling the easy connection to other related applications, e.g., CAD programs.

From the 3D models – either archived or acquired – it is possible to launch a 3D scene change detection application and detect minor changes in the environment. The tool is highly effective in detecting minute changes, though a detected change might not be necessarily associated to a security alarm. Indeed, it is up to the system's operator (e.g., inspector, police officer, security guard) to interpret the scene and validate the change.

One of the laser scanning systems showed the capability to detect smoke plumes and map its dimension and orientation. Though it was not possible to fully represent a whole plume, one can say, however, that the acquired 3D plume information can be used as a simplified "ground-truth" representation of the plume's behaviour – dimensions and orientation. This can be relevant in fine tuning some dispersion modelling parameters.

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# Laser Item Identification System Development for a Laser Based Identification of $UF_6$ Cylinders

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***Abstract:***

*Providing Continuity of Knowledge (CoK) on feed, product and tails material contained in $UF_6$ cylinders that enter and exit enrichment plants has to date posed a substantial challenge to the IAEA. Control of the stored unsealed cylinders is also a challenge. Currently, these $UF_6$ cylinders can only be manually identified by steel ID plates – welded in place or riveted or bolted to the front of the cylinders – unable to be authenticated by any reasonable method. The IAEA is testing a laser based method that identifies individual $UF_6$ cylinders by the intrinsic spatial irregularities that are unique to each cylinder. The technique, developed by the European Joint Research Centre (JRC) in Ispra, has the potential to enhance the Agency's ability to verify $UF_6$ transfers. If proven feasible, the IAEA would incorporate this technique into unattended systems coupled with video surveillance.*

**Keywords:** 3D laser surface scanning; unique identification; authentication; $UF_6$ cylinders

## 1. Introduction

The IAEA safeguards (SG) criteria require that nuclear material remains under continuity of knowledge, but also that it can be verified for accountancy during physical inventory. Providing Continuity of Knowledge (CoK) on feed, product and tails material stored in $UF_6$ cylinders that enter and exit enrichment plants has to date posed a substantial challenge to the IAEA. Control of the stored, unsealed cylinders is also a challenge.

Identification is one of the components required to be part of a surveillance module which also includes weighing scales and camera surveillance. Currently, $UF_6$ cylinders can only be manually identified by steel ID plates, welded in place or riveted or bolted to the front of the cylinders, and cannot be authenticated by any reasonable method. The amount of work required to clearly identify and account the cylinders is substantial since the IAEA inspector would have to maintain a double accountancy: the facility identification code (no possible trusted authentication of the paper labelling) and the manufacturer labelling of the cylinder.

Using the existing surveillance cameras turned out not to be feasible since their resolution is too low in order to allow a unique identification. A second approach pursued by the IAEA was the use of dedicated unique identification tags (LBIMS project). However, this approach also had a number of significant drawbacks: Firstly, tampering with the tags was an obvious concern. Secondly, it turned out to be very difficult to find a fast, reliable and cost-efficient way to permanently attach the tags to the cylinders.

Therefore, in order to make identification and authentication of $UF_6$ cylinders more reliable and efficient, the IAEA is now testing a "Laser Item Identification System" (L2IS). The system uses three-dimensional laser scanning of the cylinder faces in order to identify and authenticate individual $UF_6$ cylinders by the intrinsic, three-dimensional spatial characteristics and irregularities of the surface of each cylinder. The system is under development at the Joint Research Centre (JRC) of the European Commission in Ispra, Italy.

The use of three-dimensional laser scanning in the context of unique identification of nuclear containers has been proposed by Bovisio et al. [1]. Laser scanning is also used for nuclear containment verification [2, 3] as well as the verification of design information [3, 4, 5]. A summary of other laser-based technologies in the context of nuclear safeguards, including spectroscopic methods, optically stimulated luminescence and Lidar, can be found in [6]. The L2IS system discussed in this paper is based on laser triangulation which is illustrated in Figure 1: A sheet of light is projected onto the object using a laser diode and a cylindrical lens. Where this sheet of light intersects with the object surface, it creates a laser line which is viewed by a digital camera from a different angle. Typically, the camera is equipped with a bandpass filter adapted to the laser wavelength such as to minimize the influence of ambient light. The shape of the laser line "seen" by the camera depends on the shape of the object. Assuming the system is properly calibrated, each point of the laser line recorded by the camera yields the coordinates of one point in the laser plane. Therefore, each camera image yields a profile which is the intersection between the laser plane and the object surface. By moving the object or the scanner in a controlled manner, a sequence of profiles can be acquired, yielding a dense cloud of 3D points on the object surface.

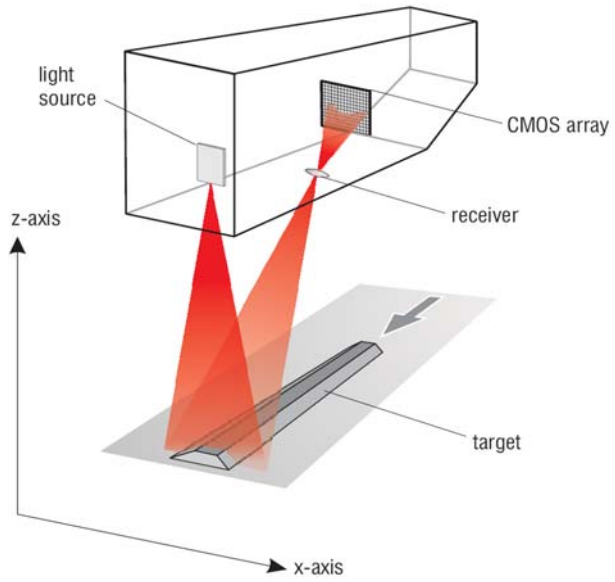


**Figure 1:** Measurement principle of laser triangulation (figure courtesy of Micro-Epsilon).

## 2. Instrumentation

A preliminary version of the measurement instrument is shown in Figure 2. The laser line scanner is mounted on a rotation table which rotates it in front of the object to be scanned. The instrument scans a field-of-view of approximately 800 x 800 x 450 mm (width x height x depth) in approximately 4.5 s. A second variant of this instrument is currently under development where the scanner remains stationary while the cylinder to be scanned is moved by the scanner on a rail.

The laser operates in the visible spectrum (660 nm) and has an output power of approximately 60 mW. It is classified as class 3B. An alternative scanner with an output power of 16 mW and laser class 2M was also tested in the field. However, it yielded unsatisfactory results on cylinders which were either painted with dark colours or completely unpainted. It should be noted that laser triangulation relies on the diffuse ("Lambertian") scattering of the laser light on the surface; similar to the scatter that occurs on a sheet of paper. It is approximately direction-independent, as opposed to the specular

reflection which dominates on surfaces such as unpainted, polished metal. Therefore, shiny metallic surfaces are – together with completely black surfaces – the worst case scenario for laser triangulation because the laser line will appear extremely dark to the camera, except if the surface inclination happens to be at the angle of specular reflection between the laser line and the camera.

**Figure 2:** Prototype instrument.

An example of a 3D model acquired with the instrument is shown in Figure 3. It shows the result of scanning a mock-up of a $UF_6$ cylinder. The ID plate can be clearly seen, as well as a small, circular marker which we have used to imitate the cylinder valve. The hole in the centre was necessary for manufacturing the mock-up, but is obviously not present in real cylinders. The figure also shows the structure of the weld seam between the outer cylinder surface and the cylinder front. Parts of the outer collar which protects the cylinder valve against physical damage are not present in the 3D model, partly due to occlusion effects, partly because they were outside the field-of-view. This, however, has no impact on the system performance, as long as the majority of the collar remains visible. The colouring has been added to the figure to convey a better three-dimensional impression. In the original model, however, the brightness or each point corresponds to the reflectivity of the surface as seen from the respective camera angle. In the next section, we will describe the processing steps necessary in order to uniquely identify a $UF_6$ cylinder using such a 3D model.

## 2. Image Processing Algorithms

### 2.1. Calibration and 3D Reconstruction

This first step is necessary even before a model such as shown in Figure 3 is available. Note that each profile acquired by the rotating laser line scanner needs to be correctly remapped into a common 3D coordinate frame. This can be done if the position of the rotation table at the time of the profile acquisition is known, as well as the exact position and orientation of the rotation axis with respect to the internal coordinate system of the scanner. Since it is not realistic to mechanically measure the position and orientation of the rotation axis on the one hand and of the scanner coordinate system on the other hand at the sub-millimetre accuracy required, we have proposed an auto-calibration scheme in [2]. It is based on acquiring several scans from different viewpoints of a planar calibration target which is known to have a very high planarity (e.g. an optical breadboard or a coated glass pane). The sought calibration parameters are then numerically computed in such a way that the planarity of the 3D-reconstructions of the calibration target is maximized. We have shown that with this method it is possible to accurately calibrate the system and to obtain true 3D surface maps.

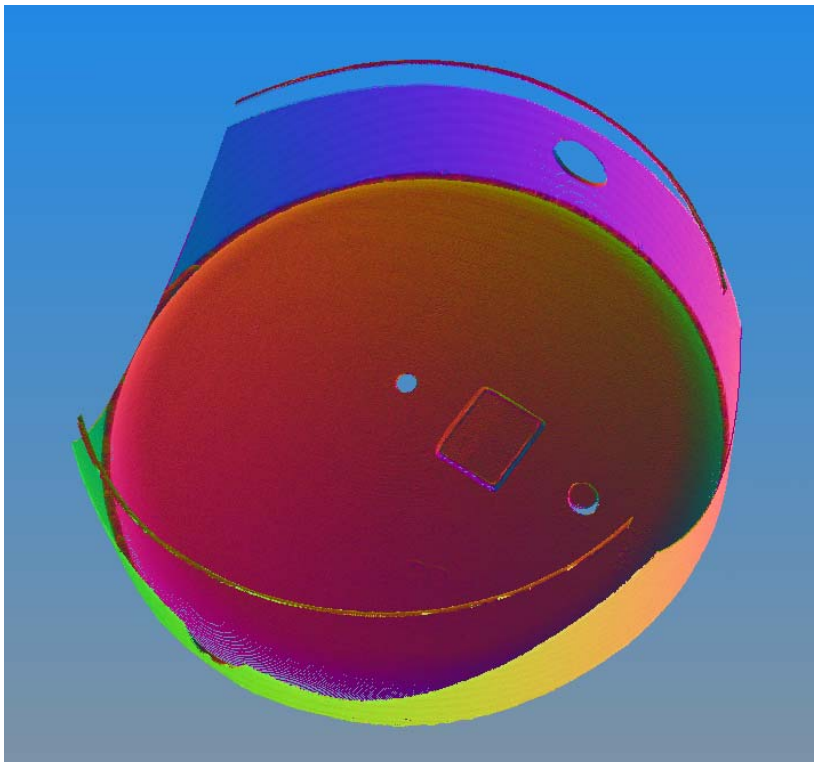

**Figure 3:** 3D laser scan of the front of one of the $UF_6$ cylinder mock-ups. The ID plate can be clearly seen, as well as a small, circular marker which we have used to imitate the cylinder valve.

## 2.2. Normalization

Now we have a true 3D surface map as the one shown in Figure 3, but the position and orientation of the cylinder front is still arbitrary because for each scan, the relative position and orientation between the scanner and the cylinder may be different. Therefore, as a second processing step, the surface needs to be normalized in such a way that a direct comparison between two scans becomes possible. We do this by identifying points on the rim of the cylinder front. Ideally, the rim would form a circle. We use the robust fitting algorithm RANSAC [7] to fit a parametric model of a circle in 3D to the identified rim points. The RANSAC algorithm is robust with respect to outliers, such as points which have been erroneously identified as rim points. The circle radius can be used for a first sanity check and for determining the type of cylinder (30B, 48Y) because the rim radii of the different cylinder types are known in advance. If the radius is accepted, then we apply a rigid 3D transformation to the scan in such a way that the circle centre is transformed into the origin of the coordinate system and the circle gets to lie in the x-y-plane. This is what we refer to as a “normalized” representation of the scan.

Figure 4 shows the normalized representation of the scan from Figure 3. The collar has been removed because it is not used for the subsequent inspection and identification steps. The rim of the collar would form a circle in the x-y-plane, its centre being in the origin of the coordinate frame. Therefore, the surface of the cylinder front is parallel to the x-y-plane but is lying below this plane. By now we have made the scan independent of the scanner position and orientation with respect to the cylinder, except for one last degree-of-freedom which is a rotation about the z-axis. This will be tackled in the next processing step.

It should be noted that when creating a new reference, the steps described in Sections 2.1 (3D reconstruction) and 2.2 (Normalization) only need to be computed once and the results can be stored. When matching a new scan against a set of references, only the new scan needs to be processed.

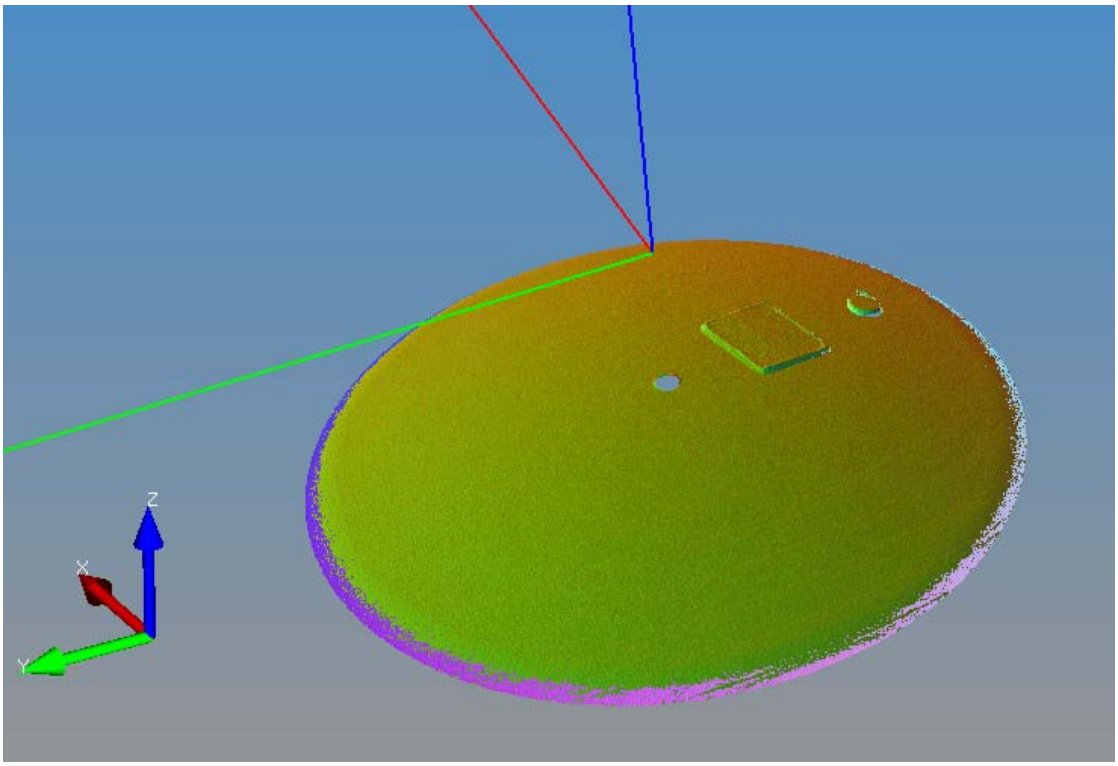


**Figure 4:** The scan from Figure 3 after normalization. The outer collar of the cylinder which protects the valve against physical damage has been removed in this normalized representation. The rim of the collar would form a circle in the x-y-plane, its centre being in the origin of the coordinate frame. Note that the surface of the cylinder front is below the origin by the height of the collar. Except for a rotation about the z-axis, the 3D-pose of the cylinder front is uniquely determined by the normalization.

## 2.3. Rotation Recovery

Obviously, if the surface of the cylinder front were perfectly symmetric, there would be no way to determine its rotation orientation. In practice, however, it is not; the most prevalent non-symmetric feature being the valve on the cylinder front. In a first approach, we identified the position of the valve in the normalized scan and then rotated the scan about the z-axis such as to align the valves in the two scans to be compared.

From an operational perspective, however, the cylinders are intended to be monitored in locations – during entering or leaving the facility on a trolley – where a valve cover will be present and will occlude the valve. It must be noted that valve covers get exchanged frequently and are manually fixated, so they cannot be assumed to maintain their 3D shape and position between scans. Therefore, we had to modify our approach such as to completely exclude the area of the valve and also the entire area where a valve cover might be present.

For the rotational recovery, there were no other features left which are known to exist on each cylinder front in the same place under any circumstances. While most cylinders have a steel ID plate welded or bolted to their front, the sizes and positions of these plates may vary widely. We therefore chose an approach which makes no *a priori* assumptions about which structure is expected on the cylinder front, as long as there is some structure present which is not rotationally symmetric. It turns out that this is always the case: In most cases, the ID plate will be the most prominent feature, but even in scans of completely symmetric surfaces without any features, we found that manufacturing irregularities and asymmetries are in most cases sufficient to recover the correct rotation angle.

Our approach is based on extracting a number of concentric, circular profiles from the normalized scan, as shown in Figure 5. The figure shows the profiles extracted from a scan to be identified (red) as well those from the respective reference scan (blue). A total of 10 circular profiles were extracted at the radii $i \cdot 23.5$ mm, $i = 1, 2, \ldots, 10$. The ID plate can be clearly seen in several of the profiles from both scans. It serves as the main cue for the rotational recovery. In the red profiles, the valve imitate is

also present, but it has been removed in the reference scan because as discussed above the entire area of the valve and any potential valve covers cannot be used.

For each of these profiles, the periodic cross-correlation function is computed between the two scans to be compared. For many of the profiles, there will be no structure present, i.e. the profiles are completely flat and so is the cross-correlation function. For those profiles, however, where exhibit unique features, the cross-correlation function will take on its maximum at the rotation angle at which the two profiles match best. This maximum, however, may not be very pronounced and reliable. In order to get a robust estimation of the correct rotation angle, we multiplicatively combine the cross-correlation functions over all concentric profiles. Profiles without structure will not change the results, while profiles with characteristic structure contribute to the overall cross-correlation function. As shown in Figure 6, the resulting overall cross-correlation function shows a pronounced maximum at the correct rotation angle.

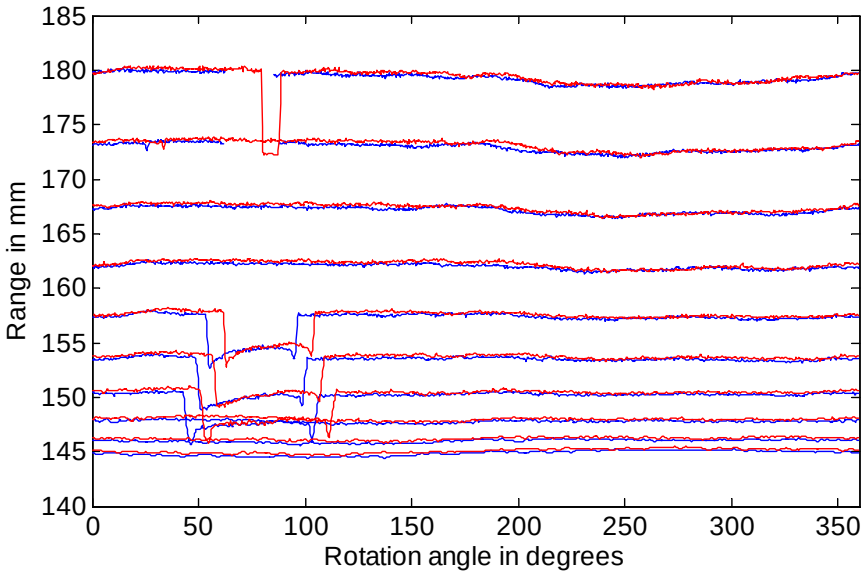


**Figure 5:** A number of circular range profiles extracted from two scans. The profiles from the scan to be identified are red; the profiles from the reference scan are blue. The ID plate can be clearly seen in the profiles. It serves as the main cue for the rotational recovery. The valve imitate can be seen in one of the red profiles, but it is absent in the reference profiles because the entire valve area is ignored in the reference scans.

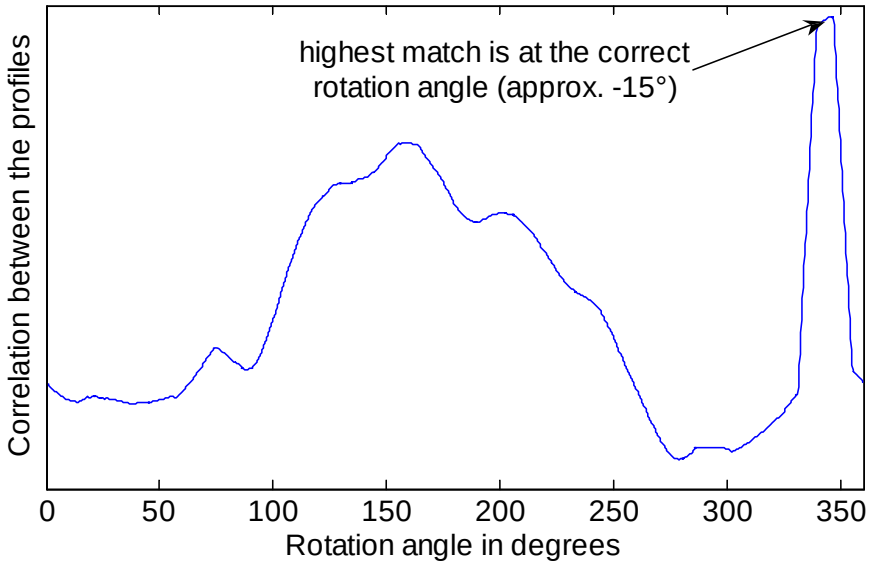


**Figure 6:** The plot shows the multiplicative combination of the correlations between the profiles from Figure 5. The correlation is plotted on a logarithmic scale. The maximum occurs at approximately 345°, i.e. the scan to be identified needs to be rotated by -15° in order to make it match optimally with the given reference scan.

## 2.3. Identification and Authentication

Knowing the correct rotation angle, the scan to be matched against a reference scan can be rotated by this angle such that the two scans can be directly compared to each other. We will refer to this comparison of an "inspection" of a new scan against a given reference scan.

The task of unique identification and authentication then consists in matching a given scan against all available references. Note that there are three conceivable outcomes of this exercise:

1. Exactly one of the inspections yields a match; all others no do match: This is the (desirable) case of a unique identification. The cylinder is concluded to be identical with the reference it matches against.

2. More than one of the inspections yield a match: In this case, there is ambiguity. It might be resolved using other available data. In general, however, this result will raise an "alarm" situation, i.e. the inspector will have to manually review this event more thoroughly.

3. None of the inspections yields a match: Either the cylinder really was not identical with any of the available reference cylinders, or the system has missed the correct reference. In either case, this is an "alarm" situation which will have to be manually reviewed.

It remains to be shown how a single scan is inspected against a single reference: After normalization and recovery of the z-rotation angle, we can compute point-to-surface distances between each point in the new scan to the surface of the reference scan and examine the distribution (histogram) of the found differences. If the two surfaces were identical, then we would expect the pixel-by-pixel differences to be normally distributed, the variance of the normal distribution being twice the variance of the scanning instrument as such (the factor two results from the computation of the difference). Our decision whether or not we will accept an inspection as matching or not, is based on a weighted percentage of pixels in the difference image which lie outside of this expected normal distribution. Experimentally, we found a threshold of 1% to work well in practice, i.e. if no more than 1% (after weighing with the absolute value of the difference) of the pixels lie outside of what we consider the normal instrument uncertainty, then we will accept the inspection as matching.

Figures 7 and 8 show the inspection of a scan against two reference, one matching and one non-matching. The colour of each point represents the point-to-surface distance on a scale from 0 mm (dark blue) to ±1 mm or more (dark red). Figure 7 shows the situation of a matching reference: All pixels are in the order of measurement noise of the instrument (dark to light blue), except for the weld of the ID plate which shows larger deviations due to residual errors in the normalization and rotation recovery steps. Note how the area around the valve was ignored in the inspection (dark blue rectangular area in Figure 7). Figure 8 shows a non-matching example: Even though the two cylinder fronts visually look indiscernible to a human observer, there are now much stronger deviations which cannot be explained by measurement uncertainties. Yellow and red areas are present both on the ID plate and across the cylinder surface.

The distinction between a matching and a non-matching scan can be quickly made by visually inspecting images like the ones in Figures 7 and 8. In a practical setting, however, we need to automate this decision since each scan needs to be inspected against a large number of references. To this end, we examine the cumulative density function (cdf) of the point-to-surface distances and compare it to a theoretically expected cdf of a normal distribution. Figures 9 and 10 show the results for the examples from Figures 7 and 8. The red lines are the theoretically expected cdf's and are identical in both figures. The blue lines are the actually observed cdf's. In Figure 9, the blue line lies above (to the left) of the red line almost everywhere. This means that the point-to-surface distances are even lower than one would accept based on the instrument noise. In Figure 10, there is an extended area where the blue line lies below (to the right) of the red line. This area corresponds to distances which lie outside of the measurement uncertainty. The cdf can be clearly rejected as non-matching based on this criterion.

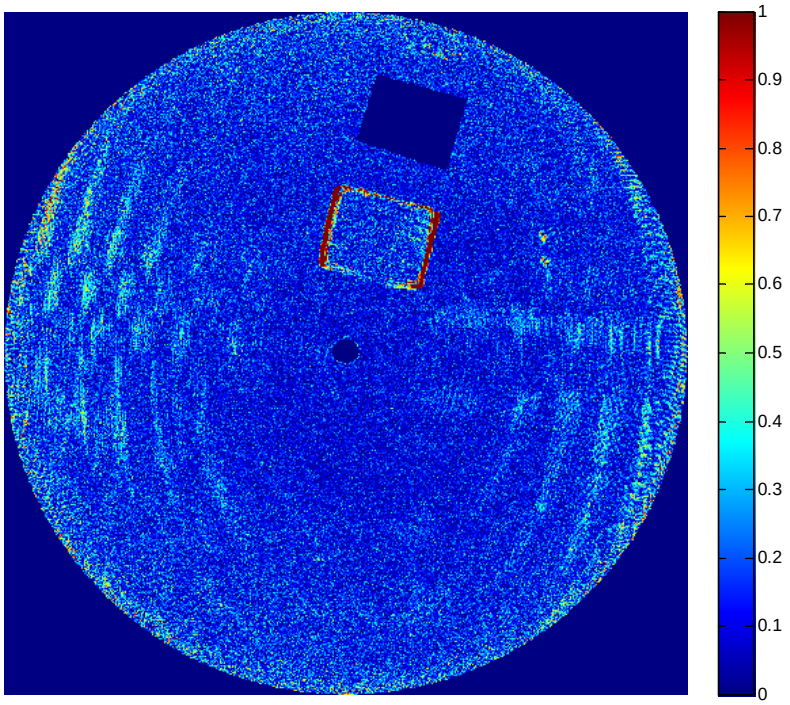


**Figure 7:** Inspection of a scan against a matching reference. All distances are in the order of magnitude of the measurement uncertainty of the instrument (dark to light blue), expect for the weld of the ID plate where there are some larger deviations due to residual errors in the normalization and rotation recovery.

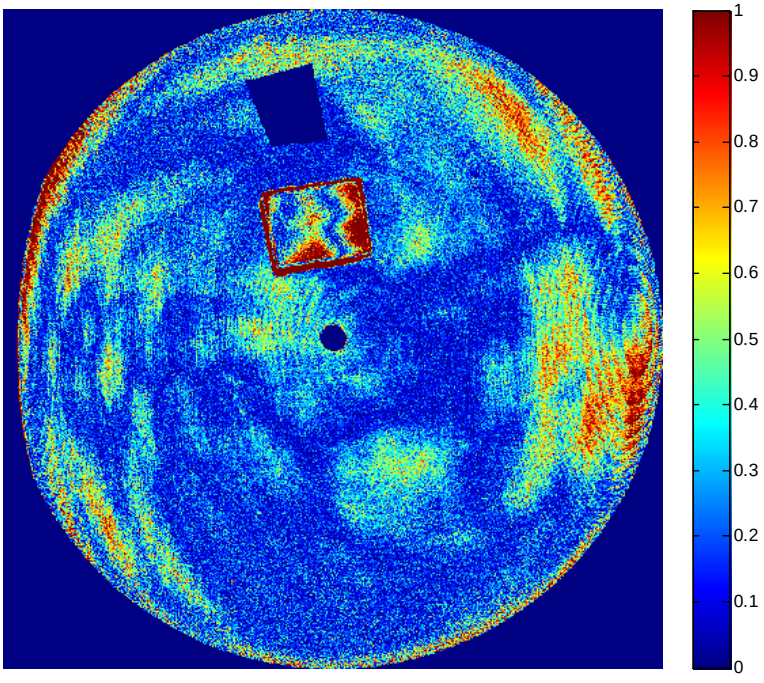


**Figure 8:** Inspection of the same scan as in Figure 7, but against a non-matching reference. It can be clearly seen that there are differences, both on the ID plate and on the cylinder surface as such, which cannot be explained by instrument or normalization uncertainties. These differences are true deviations in the 3D surface structure due to manufacturing tolerances.

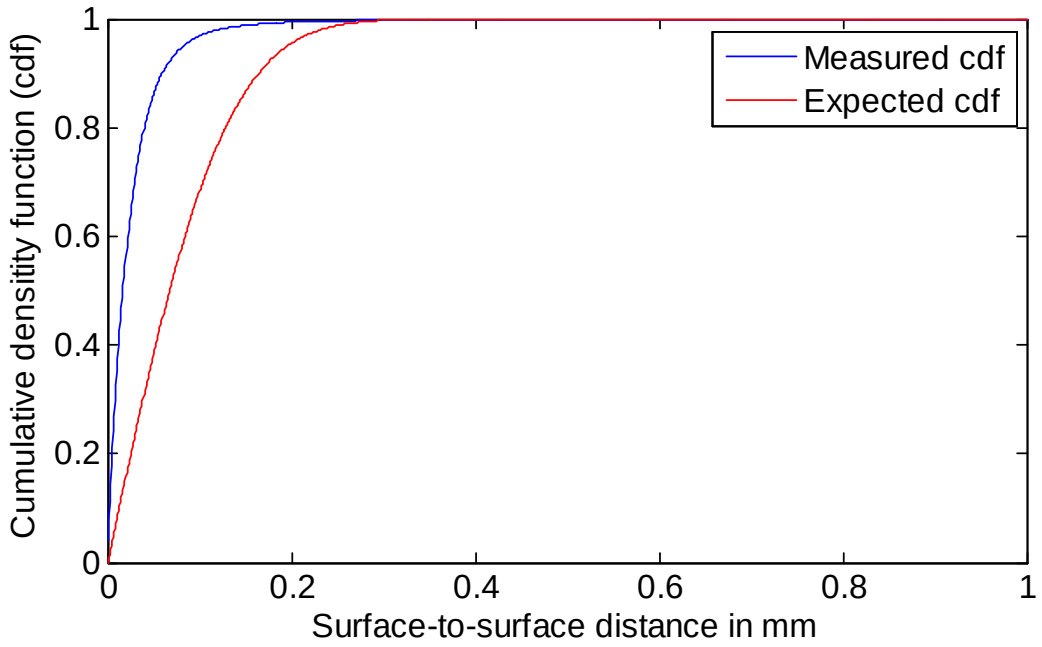


**Figure 9:** Distribution of the surface-to-surface distances for the matching inspection from Figure 7. The distributions are shown as cumulative density functions (cdf), i.e. a value of 0.4 at 0.05 mm would mean that 40% of the pixels in Figure 7 have a surface-to-surface distance of 0.05 mm or less. The red line is a theoretically computed cdf for a Gaussian distribution with a variance as would be expected for the overall system. The blue line is the measured cdf for Figure 7. The blue line lies above the red line everywhere which means that all surface-to-surface distances can be explained by the expected instrument uncertainties. In fact, the blue line is even slightly better than what could be expected from the instrument.

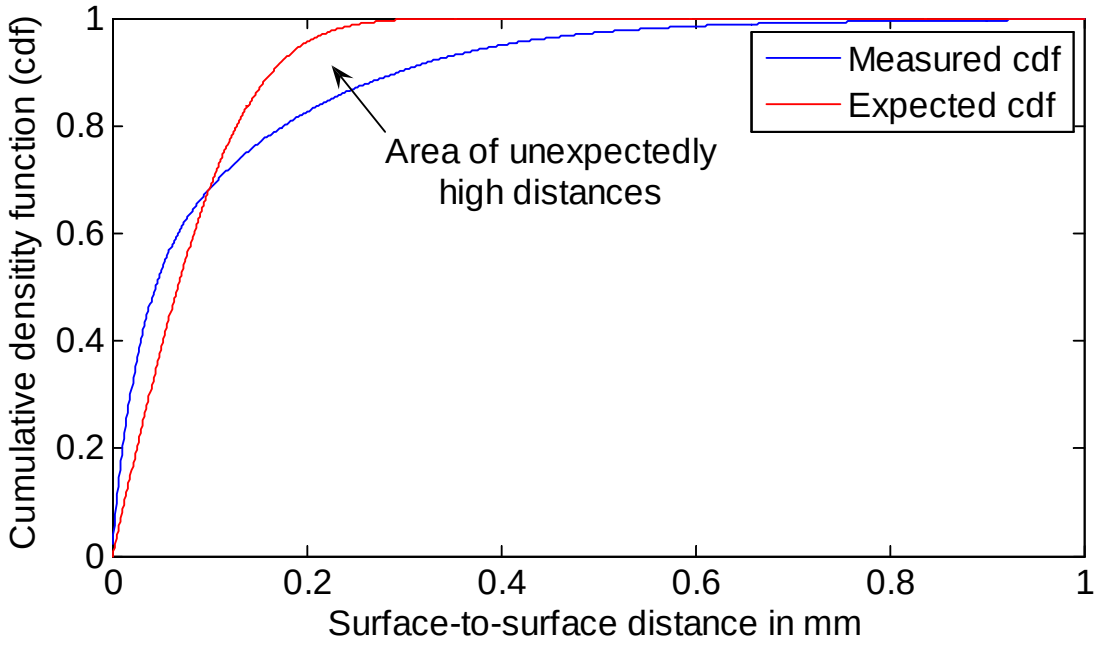


**Figure 10:** Distribution of the surface-to-surface distances for the non-matching inspection from Figure 8. Above approximately 0.1 mm, the blue line is clearly below the red line. The area between the two lines (blue below, red above) is a measure of how strongly the surface-to-surface distances are beyond the deviations which can be explained by the measurement uncertainty. In this case, the inspection can be clearly rejected as non-matching.

## 3. System Development Roadmap

The first feasibility demonstration took place at the JRC in Ispra in September 2006 and was performed on mock-ups of $UF_6$ cylinders. Using four full-scale front and end surfaces of $UF_6$ cylinders it was demonstrated that the system in principle is capable of distinguishing between the four surfaces even though they exhibited no intentional differences, not even differences visible to the human eye.

In December 2006, the L2IS project entered a field test phase: Two systems, based on similar hardware but with different laser powers were brought in the field and tested on real $UF_6$ cylinders in a facility. As detailed above, only the system with the higher laser power yielded satisfactory results. After acquiring a number of indoor scans on both 30B and 48Y cylinders, the class 3B laser system was installed indoors and outdoors. As a side effect, the weather conditions allowed us to verify that the system works reliably under heavy and persistent rain. Moisture on the cylinder surfaces does not have a negative impact on the quality of the scans.

Another important lesson learned from the field trial was that in a realistic setting, valve covers may be present and their exact shape and position may change from one scan to the next. This led to the modifications in the rotation recovery detailed above. With these modifications, the field test brought very positive conclusions. In almost all cases, a correct and unique identification of a cylinder front was possible. In some pathological cases (surfaces without any structure such as ID plates, extreme positions and orientations of the scanning instrument) false alarms were generated in the sense that either the cylinder could not be matched at all against any of the reference or in the sense that the identification yielded an ambiguous result (multiple matches). However, there was no situation where a unique, but incorrect identification was made.

A second field test phase is foreseen for 2007. Its main goals are to get more experience with 48Y cylinders and to acquire data for the stationary scanner with the cylinder being moved along the scanner on a trolley on rails. A pilot installation in a facility is also foreseen for 2007.

## 4. Conclusion and Outlook

We have demonstrated in laboratory tests as well as in a field trial that the L2IS system is capable of uniquely identifying $UF_6$ cylinders, only based on their three-dimensional surface structure with its intrinsic irregularities and manufacturing tolerances. The key advantage of the approach is that it is a true “self-authentication” technique, comparable to biometric identification techniques for humans (fingerprinting, iris scanning, etc.). It does not rely on dedicated labels, tags, markers, or the like. Attaching such dedicated identification aids is not feasible from an operational point of view and would yield a natural attack point.

The IAEA has high expectations regarding the L2IS system as it will provide automated identification of transfers to and from process areas, provide recorded inventory control, and enhance the Continuity of Knowledge at enrichment plants with a new perspective. Potentially, the number of inspection days for these plants could be reduced.

If the system fulfils the safeguards criteria requirement, the IAEA could incorporate it into unattended systems, coupled with video surveillance. In this case the system would offer a backup to the video and vice versa: The laser scanner could detect what the 2D surveillance camera cannot “see”, e.g. in case of a loss of illumination; and if a counterfeit surface was to be presented to the scanner, the video would raise an alarm. Coupling the system with multiple installed video systems could offer further possibilities, e.g. in terms of sharing data storage or power supply. A remote monitoring would be the next step ahead.

## Acknowledgment

The authors would like to express their appreciation to Mr. David Puig from the JRC for his hard work and dedication in the mechanical design and assembly of the system and his support of the field trial.

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# Further Development of Laser System for Design Verification Inside and Outside Nuclear Facilities

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***Abstract:***

*Through a European-Commission Support Program Task, the IAEA has successfully implemented a 3-Dimensional Laser Range Finder (3DLR) that has demonstrated the ability to accurately map the interiors of complex facilities, such as the Reprocessing Plant at Rokkasho, for the purpose of performing design information verification (DIV). The IAEA envisions the need for a similar laser based system to perform DIV activities outside and around complex nuclear facilities. A similar laser-based system has been developed and is capable of providing detailed design information for large complex facilities that have an array of buildings and features. In addition, the system has software that combines the highly accurate spatial data of this mobile ground based laser with satellite imagery and/or high-resolution digital photography. The result is a spatially accurate image that not only is useful for DIV but also for the training and preparation of Inspectors.*

**Keywords:** 3Dlaser Scanner, Accurate Model, DIV, Cost effective

## 1. Introduction

The International Atomic Energy Agency (IAEA) is using a 3-Dimensional Laser Range Finder (3DLR) system for design information verification (DIV) purposes to accurately map interiors of complex nuclear facilities under safeguards [1]. DIV activities are performed during the complete life cycle of a nuclear facility to confirm that it operates as declared and is not misused for undeclared nuclear activities.

The 3DLR system is composed from an off-the-shelf laser range finder and a specific software package developed by the JRC to create 3D models that could validate the actual “as-built” components versus the design provided in the design information. The 3D models are obtained by scanning different positions for the same area of interest. The indoor version of the 3DLR system offers millimetre accuracy for a 50 m distance at maximum scanning speed. The current technique combines the 3D model image with information from other types of sensors such as colour (see Figure 1), radiation [2] and thermal sensors to further enhance the models of the facilities. The enhanced models serve as base line for subsequent verification scans to detect any structural changes or operational conditions which might indicate a misuse of the facility.

In the same line of efficiency and verification potential, the IAEA envisages the need for a similar system to document a nuclear site and to perform DIV activities outside and around complex facilities. The range and accuracy are different from the indoor 3DLR system but indeed offer the same possibility for construction of 3D models based on several scans that are geo-tagged and could later be mapped with actual photographs. The system software could also be used for combining/ linking the

highly accurate spatial data acquired by this mobile ground based laser with satellite imagery. The possibilities offered by the obtained model would exceed the DIV scope by offering a model of facilities used for accurate documentation as well as for the training and preparation of inspectors.

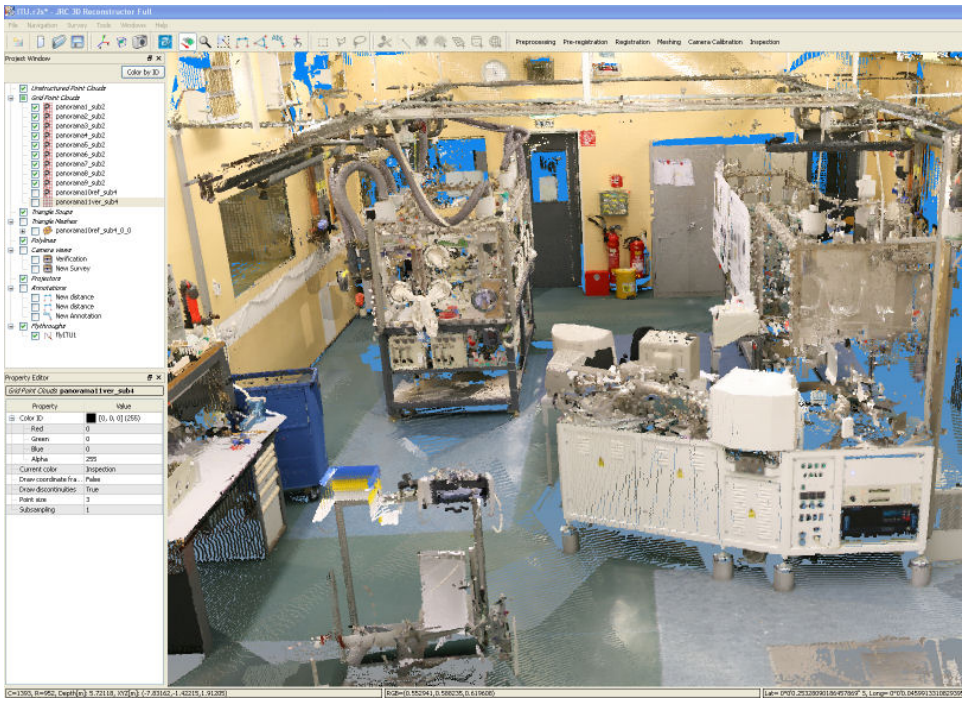

**Figure 1:** Snapshot of a 3D model from a laboratory reconstructed from 9 range images and 63 colour digital images. Acquisition time: 3hours; processing time: 4 hours.

## 2. System Description

The basic measurement devices in both the 3DLR and OVS systems are a portable commercial off-the-shelf laser range scanner and a portable computer. The laser range scanner is composed of a laser beam, a rotating mirror and a rotating base.


(a) (b)

**Figure 2:** (a) Laser scanner on tripod with dolly for indoor operation (b) Laser scanner mounted on vehicle for outdoor operation.

The laser beam is deflected in two dimensions: vertically by the rotating deflection mirror and horizontally by the rotating movement of the entire measuring device. For DIV indoor operations, we need a highly portable system with a high level of autonomy. Such a system is based on the above mentioned components mounted on a tripod with a dolly, see Figure 2a. Multiple scan-positions are generally

needed to model an environment to resolve object occlusions. The movement of the scanner is normally achieved by physically moving the tripod to suitable positions by the inspector(s). For flexibility of operation the system is battery operated and is fully self contained.

In case of outdoor scenarios the area to be covered increases by large magnitudes. In this case, one can make use of a vehicle for moving the scanner; see Figure 2b. In order to facilitate scanning and further processing steps some extra devices are present on top of the vehicle scanning platform. The different components acquire concurrently the current position, orientation and colour-information along the scanning direction as described below [3]:

- *Position:* For position estimate, a Real Time Kinematics (RTK)-enabled GPS is used. The RTK technique is based on the use of the carrier phase measurements of the GPS (or Galileo) signals where a single reference station provides the real-time corrections up to centimetre level of accuracy. The RTK system maintains a real-time package based communication with an RTK base station in the vicinity. Alternatively, the raw information is locally saved for further correction when a RTK base-station log is attached.

- *Heading:* In order to know the vehicle attitude (roll and pitch) and heading an inertial system running as an inertial measurement unit (IMU) is used.

- *Colour acquisition:* To acquire the natural texture a high resolution digital camera mounted on top of the scanner is used. This assumes a stop-scan-and-go acquisition mode (as opposed to continuous scanning). The camera is directly connected to the scanning computer for automated image acquisition. In the case of continuous scan-while-driving mode, the colour information needs to be constantly acquired. In this situation a high-resolution digital still-image camera is less useful due to limited acquisition frequency. For this scanning mode a video-camera with a wide-angle lens is used giving continuous colour information as needed to colorize the scan information.

The system is compatible with two complementary laser scanners. The laser scanners are easily swapped for different purposes. Indeed mounting the laser base on the top of the vehicle takes only a few minutes:

- For detailed analysis requiring the highest level of spatial resolution and millimetre accuracy, the system uses scanners with phase-based amplitude modulation. These scanners can reach up until 53.5/79 metres on descent target reflectivity.

- If there is a need for further distances, a true time-of-flight (TOF) based device is used. The strength of the TOF technique is the potential to detect the reflecting echo from the short, strong pulses emitted from the scanning laser. The TOF-based scanning device can perform range measurements up to 1000 metres with centimetre accuracy.

## 3. Vienna International Centre

The OVS prototype was demonstrated in November 2006 at IAEA headquarters in Vienna. The system was successfully used to create a dimensionally accurate and photo-realistic 3D model of the IAEA complex of buildings. Figure 3 shows a snapshot of the resulting 3D model and Figure 4 the detection of changes in a few selected areas. The 3D model was reconstructed in one day from 11 range scans (6 scans using a tripod: 2 acquired at the ground level and 4 from the top of two buildings; and 5 scans with the scanner mounted on the vehicle), and 77 colour digital images (7 per range scan). The digital camera mounted on the laser scanner takes pictures at 40 degree interval that are automatically matched with the laser reflectance image and overlaid on top of the 3D model (see Figure 5). The total data acquisition time was 5h30min. This includes all the time to move the scanning equipment to different positions including taking elevators to the top of the buildings, mounting the scanner on top of the car, etc. The total processing time was 5 hours. The processing equipment used was mainly two laptop PCs with the full suite of JRC's 3D Reconstruction software.

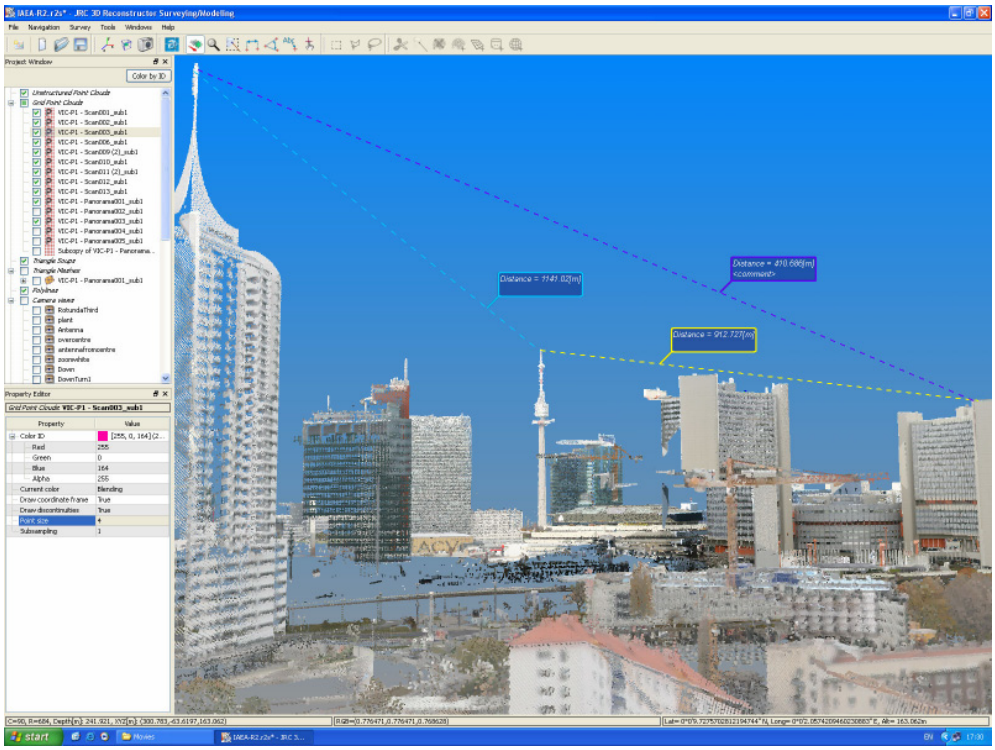

**Figure 3:** Snapshots of the Vienna International Centre 3D model reconstructed from 11 range scans.

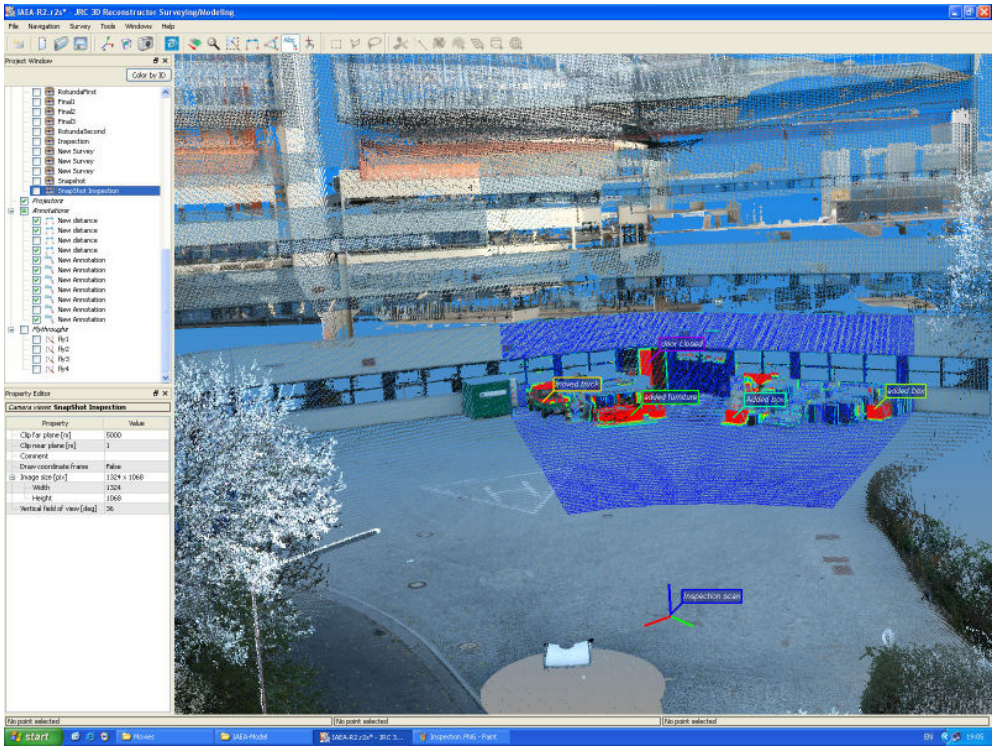

**Figure 4:** Snapshots of the Vienna International Centre 3D model with highlighted differences.

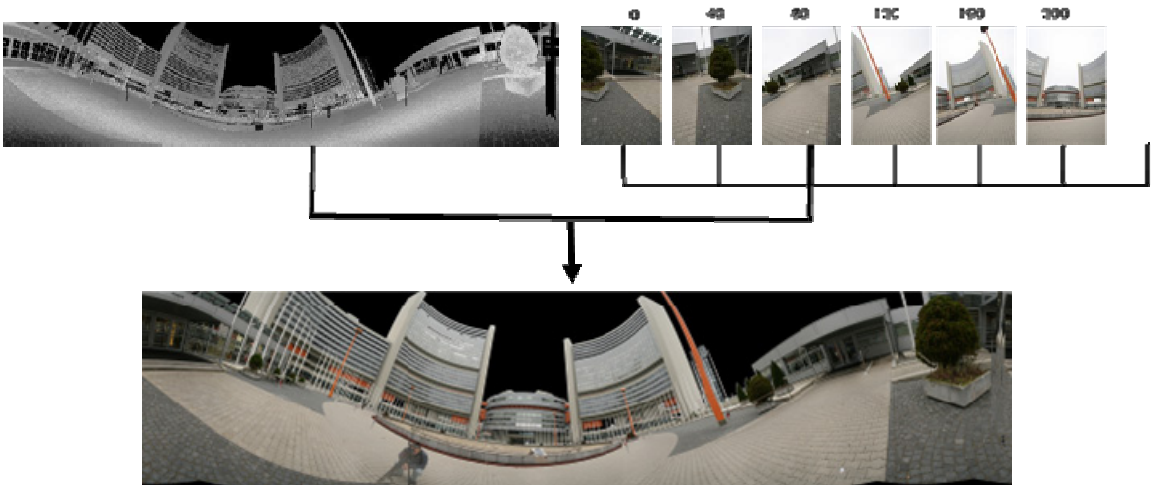

**Figure 5:** 3D colour mapping.

## 4. Conclusions

The technical principles behind the indoors Design Information Verification – DIV –system have been applied to designing and implementing a similar system for outdoors applications. The Outdoor Verification System – OVS – enables and contributes to a new Safeguards approach for facilities either already under or entering IAEA's Safeguards.

OVS's output is relevant for:

(a) Accurate documentation of a nuclear site or facility "as-is"
(b) Detection of changes – DIV – outside and around complex facilities
(c) Training of inspectors and Preparation of inspections
(d) Efficient and intuitive presentation tool for the Safeguards measurements associated to a given facility

The demonstration at Vienna's International Centre showed that the system is highly portable and can be deployed with minimum notice. Further it also showed what a team of two people can do in a few hours, both in terms of (a) data acquisition and (b) data integration and processing, including the detection of changes.

The OVS concept is easily adaptable to new and evolving hardware with improved features. Indeed, the functional core of the system relies upon its architecture and the processing software.

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# New Ultrasonic Sealing Systems for CANDU spent fuel bundles

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## *Abstract:*

*A new sealing system for CANDU reactor spent fuel bundles, to replace the AECL ARC seal, was developed, according to the IAEA requirements.*
*The new bolt for underwater sealing is derived from the design of the sealing bolts already used in the La Hague reprocessing plant. The design was revisited in order to comply with the CANDU interface requirements.*
*A first series of 15 bolts, with the new reading head and data acquisition system, was produced. Field trials began in June 2006 at Cernavoda NPP (Romania).*
*A second series of 50 bolts and one reading head was built for the Vulnerability Assessment to be done by an independent laboratory.*
*This paper discusses the design of this sealing system and the initial results of both field trials.*

**Keywords**: Ultrasonic sealing bolts, underwater seals, CANDU design, Cernavoda (Romania)

## 1 Introduction

The Seals & Identification Laboratory (*SILab*) is a laboratory of the Joint Research Centre of the European Commission. *SILab* develops technologies and equipment based on ultrasonic technologies, suitable for sealing or identification of nuclear or commercial items. *SILab* is also involved in the development of methods for the identification of weapons (using both ultrasonic or RFID techniques). RFID technologies are also used to demonstrate the potential of the smart container. As a spin-off of RFID activities, *SILab* is also developing a "SEcure and SAfe MObility NETwork" to improve independence and mobility for visually impaired people.

Regarding seals for nuclear applications, *SILab* has already developed ultrasonic seals and equipments for underwater applications that are in routine use by both nuclear control agencies (IAEA and DG-TREN/ESO) in Sellafield (UK) and La Hague (F) installations.

CANDU® Reactors are manufactured by AECL (Atomic Energy of Canada Limited) and are continuously loaded (and unloaded) with fresh fuel. The exploitation license at Cernavoda requires that spent fuel bundle stacking frames be sealed. Until now, AECL supplied the ARC seal (AECL Random Coil seals) for this purpose.

On IAEA request, a study of the application of *SILab* ultrasonic seals to replace ARC seals for Cernavoda began in 2005. *SILab* ultrasonic seals present, as main their advantage, stability against time and radiation. Being purely static pieces of stainless steal they will last the life time of the stacking frames. Only the reading equipment has to be maintained.

## 2 Basic of ultrasonic seals

An ultrasonic seal is used to close a container (or a structure). The internal structure comprises a random identity and a frangible element (integrity) which breaks when an attempt is made to remove the seals.
The reading device is comprised of a transducer which generates an ultrasonic signal and senses the reflected signal. The transducer rotates above the sealing bolt recording the ultrasonic echoes reflected over a complete revolution.

The seal replaces one of the standard bolts of the container lid. The body of the seal has the same mechanical properties as a standard bolt (thread and applied torque) and is designed for each specific application.

The core of the ultrasonic seal (most right photo below) is a cylindrical assembly containing its unique identity and an integrity feature which breaks when opened. This assembly is radiation resistant and particularly reliable even in very harsh environmental conditions.


Photo 1 : Core of ultrasonic seals.

The identification feature is an assembly of several discs randomly stamped (most left photo), which are brazed together to form a univocal identity (middle left photo). The integrity is a thin metal rod which breaks when torsion or traction is applied to it (middle right photo). The identity and the integrity are then brazed together to form the core of the ultrasonic seals (most right photo). This core is then welded into the top of the seal. The bodies of the seals are designed according to each application.

Photo 2 : Custom seals designs

The seals are read using an ultrasonic device, consisting of a transducer generating an high frequency ultrasonic pulse. The sound energy propagates through the materials and when there is a discontinuity in the wave path, a part of the energy will be reflected back from the surface. The reflected wave signal is transformed into an electrical signal by the transducer and is analyzed by the analysis device.

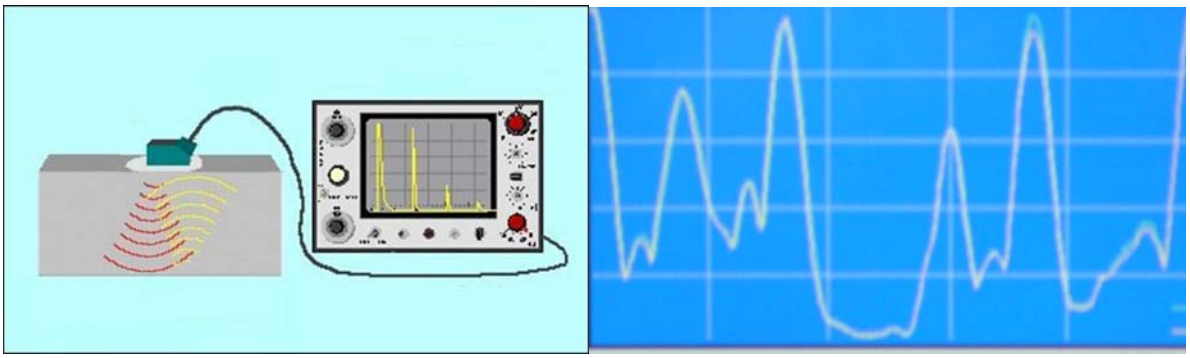

Figure 1 : Principle of ultrasonic reading and fingerprint of a seal.

In order to have a complete fingerprint of each seal, this transducer is then rotated upon the seal. In each position of the transducer the echo is then recorded. The figure below can be seen as intensity of the echo as a function of the angular position of the transducer. The X axis covered a complete revolution (360°).

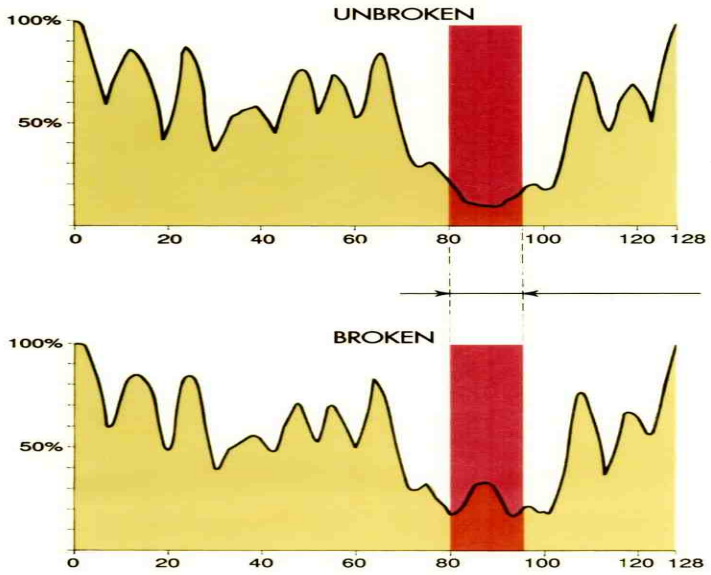


Figure 2: Unbroken versus Broken seal.

On inspections, after a complete reading, a mathematical correlation with the reference reading is calculated. The reference reading is a previous reading made by an authorized inspector at IAEA Head Quarters. The seal is considered as "Identified" if the correlation between the two readings is higher than 0.93). Then an analysis of the integrity area (the red area of the left figure of figure 2) will determine the "Broken" or "Unbroken" status.

The main advantages of these seals are that they are insensitive to radiation and they can last tens of years. Another advantage is that an inspector has an immediate answer as to the authenticity and status of the seal. In the case where these seals have are used to protect material in long term storage, the only device that would need special maintenance is the reading head.

## 3 JRC Ultrasonic seal for the underwater CANDU spent fuel storage.

A Candu reactor operates in continuous mode. New fuel is loaded and spent fuel is unloaded every day (up to 16 fuel bundles by day). The spent fuel is then carried to the spent fuel bay where it is placed in stacking frames. Once the frame is full, a frame cover and two seals are applied. The spent fuel will remain some years (typically 7 years) in the spent fuel bay until it can be stored in dry storage. Today AECL Random Coil seals (ARC seals) are used to seal the stacking frames.

### 3.1 Specifications

The need expressed by the agency was to develop a seal that can be used in place of existing ARC seals. The seal must be read with no limitation of time between two readings, giving the same result. It must also use the existing tools developed for the ARC seals.

IAEA provided an existing ARC seal and mechanical drawings of the existing tools. Fruitful discussions with AECL and IAEA inspectors were also useful to define the overall adaptations that were necessary..

When the frame cover is in place closing a stacking frame, the seal is attached by screwing it onto top of a tie-rod, which screws into a fixture at the bottom of the stacking frame. When the seal is attached, the integrity leg must be completely inserted into the grip of the tie rod.

### 3.2 The JRC Candu seal (JCS)

There were very few modifications of the exterior of the JCS seal when compared with the original AECL design. The handling and attachment of the seal were not changed.

- The cone angle was decreased from 15° to 8°. This allows better positioning of the reading head with respect to the upper part the seal.
- The distance between the hole, used to let water passed through the seal, to the centre of the seal was increased from 12 mm to 17 mm. This hole is also used to locate the position of the reading head on the seal.
- A grove is used in the AECL design to fix the location of the AECL reading head upon the seal. As the JCS seal uses the hole for this locator, the groove no longer exists.

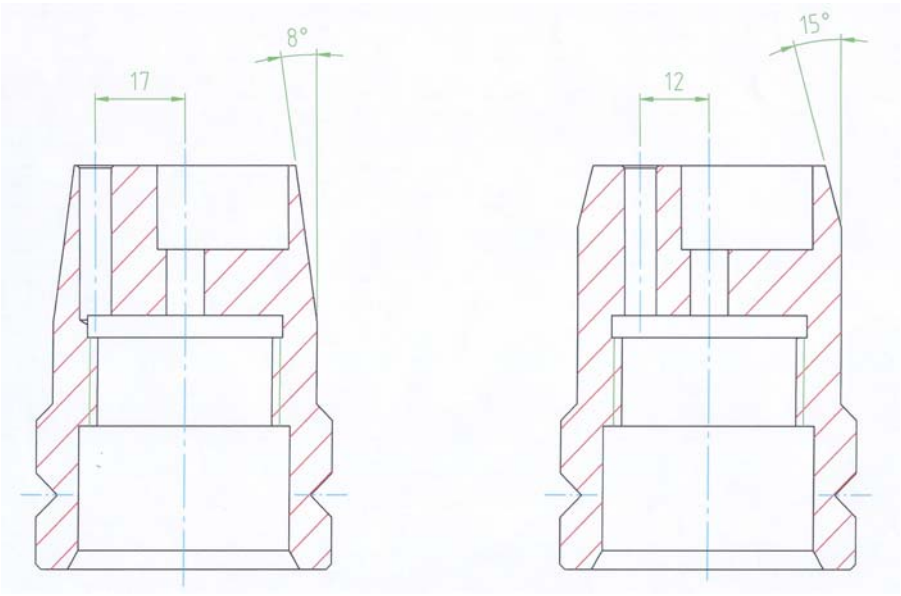


Figure 3: JRC design (left) versus AECL design (right)

When the seal is placed attached, the integrity element must completely engage with the grip of the tie rod as shown in the figure below.

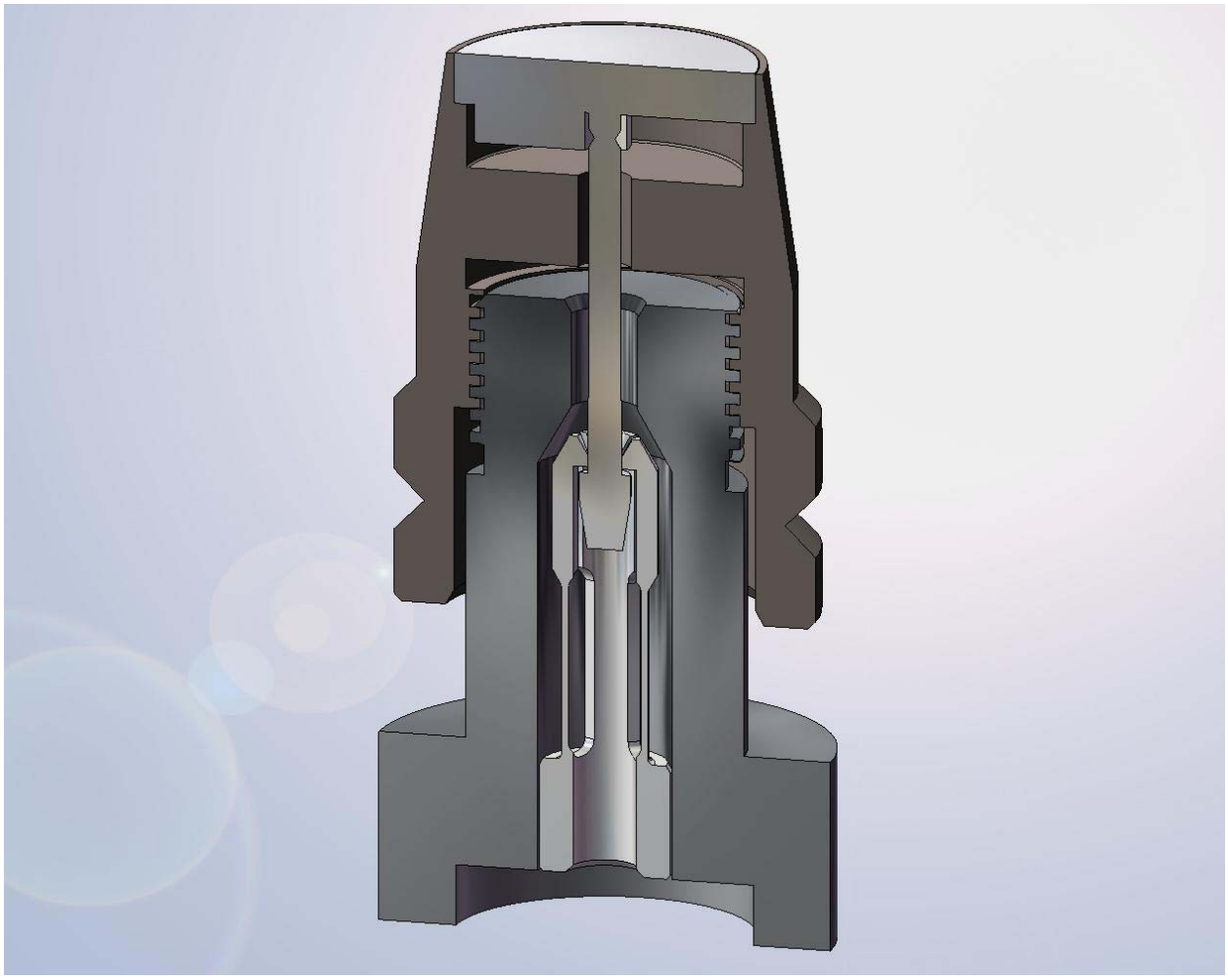

Figure 4 : Seal applied in the tie-rod

When the seal is removed, the integrity element is retained by the grip. The integrity element breaks from the upper restriction (not visible in the above figure). It then falls into the tie rod where it remains. The identity feature remains unchanged.

When a broken seal is read the reading system will detect the absence of the integrity feature but will be still able to check the identity.

### 3.3 The Reading Head

This new Reading Head is specifically designed for the reading of JCS seals. The design provides the ability to check the identity and integrity with a single measurement.

The reading head uses the same ultrasonic and motorisation modules as currently used in La Hague and Sellafield. The overall geometry has been adapted to the specific shape of JCS seals. The newly designed reading head has been successfully tested in 20 meters of water.

The mechanical interface on the top is identical to AECL reading heads to use the same tools for the handling.

Photo 3: JCS Reading-Head

The reading head uses the cone shape of the seal to help centre itself on the seal.

The hole on top of the seal is used to position the reading head. A pin of precise diameter will enter in this hole when the reading head is in the correct angular position. The pin has a conic shape to facilitate entry into in the hole. For reading, the inspector places the reading head on the seal and rotates it until he feels the pin engage. This is an easy operation even with a tool shaft length of several meters.

### 3.4 The software

The inspection software was conceived to facilitate "on site" inspections. The first menu level contains only the functions an inspector will have to perform at a facility. These functions are attaching, verifying and detaching a seal. Management functions are accessible from a second level menu.

In the management menu the inspector has the ability to make a reference reading, look at single measurement, compare two measurements and look at the status of all applied seals.

When measuring seals, the relevant parameters are displayed. These parameters are also saved with the measurement and shown when an existing measurement is displayed.

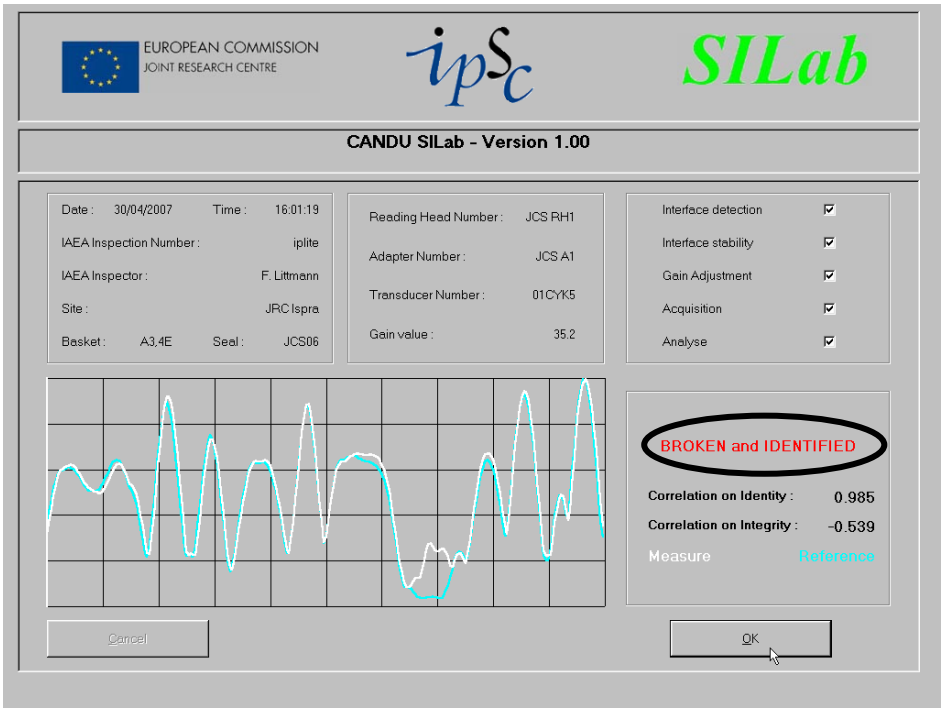


Photo 4: Reading screen of the Candu® software

After any acquisition, the reading is compared to the reference reading and the status of the comparison is indicated together with the correlation coefficient. The status of the seal is indicated in green when the result is as expected, and in red when not as expected. The example above shows an unexpected broken seal.

The curves (both reference and measured) are shown. The correlation coefficient between the reference and the newly acquired measurement are also shown. This information may be useful in case of dubious results.

Every stacking frame is sealed with two seals. By convention a stacking frame is identified by its position in a coordinate grid. The seal locations are identified by the stacking frame identifier and the letter E or W, designating that the seal is installed on the East or West side of the frame. As there is no indication of East and West in the bay, it is possible to misidentify where a seal in attached. The software takes this into account. When a seal is found as "Unidentified" in a given position, the software checks the correlation with the seal in the other position to verify the problem was not caused by misidentifying the seal location.

The software automatically prevents an inspector from placing a new seal on a frame that has already been sealed. In the case where seals have been removed from a stacking frame, the software will ask for confirmation before authorising the re-sealing of the frame. This may happen when the fuel bundles have been transferred to dry storage.

A parameter file is saved automatically after any changes in the measurement parameters. The measurements parameters are only accessible from a special menu.

# 4 Manufacture of JCS seals

## 4.1 Identity core manufacture

The cores of the seal are realized in a one step brazing process.

The identities disks are stacked in a random disposition on the integrity piece. Brazing paste is put in several parts of the stack in a quantity that will adequately bind the disks, but not fill all the holes. This assembly is then heated up to 1000°C for several minutes in the furnace.

Photo 5: Heating of an identity (left) and final identity + integrity (right).

Once produced, all identities are checked individually. As the diffusion of the brazing follows a random process, it is not possible to predict the identities that will be produced.

## 4.2 Manufacture of the seals

The bodies of the seals are machined in two steps. A first partial machining is done to rough out the seal body. Then the identities are welded on these bodies and the final machining of the seals can be done.

Photo 6: Rough body of a seal (left), welded identity of top a rough seal (center) and final machining of the seal (right)

# 5 Tests of the JCS seal

## 5.1 Test on the first batch of seals

A first batch of seals was delivered to the IAEA in February 2006. They were read at IAEA head-quarters in Vienna and then installed at Cernavoda (Romania) in March 2006. The first verification of these seals was accomplished in June 2006.

Special software was developed for these tests. This software saves data files in Excel format to perform/define more sophisticated treatment of the data. These treatments were later implemented in the inspection software.

The table below indicates the correlations between Vienna readings and Cernavoda's.

| Seal | Correlation |
|---|---|
| JCSTest001 | 0.994 |
| JCSTest002 | 0.991 |
| JCSTest003 | 0.999 |

Table 1: Correlations between Vienna and Cernavoda readings (same reading head)

A seal is considered identified if the correlation coefficient is greater than 0.93. The same reading head was used for both readings.

A seal was then broken and read again. The figure below shows the response of the test system:

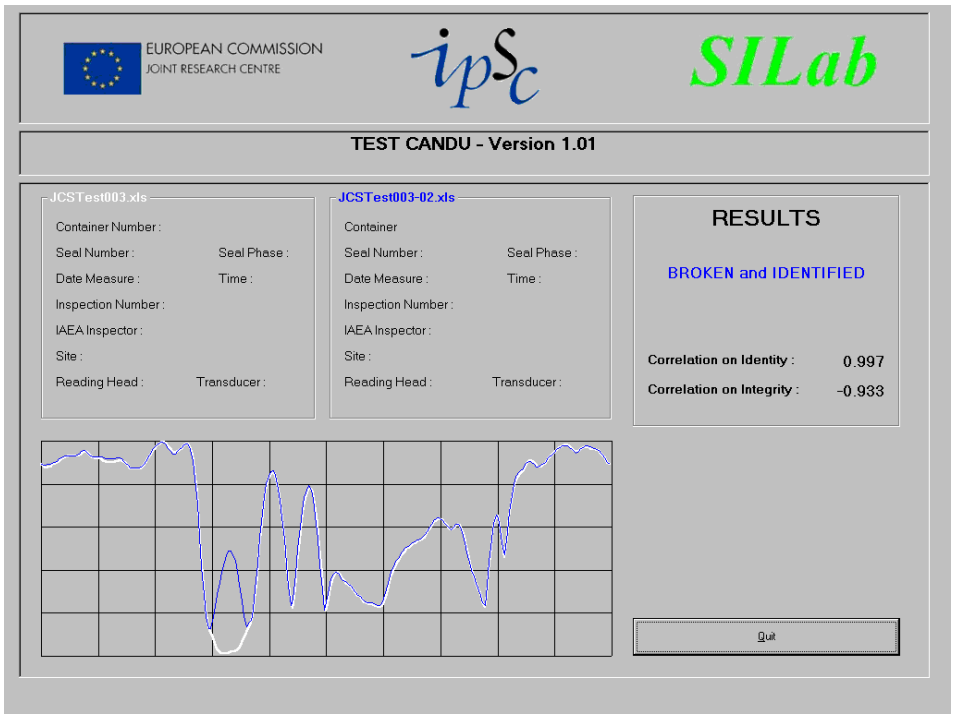


Photo 7: Reading of a broken seal in Cernavoda

The seal is perfectly identified (correlation of 0.997) and detected as "Broken".

### 5.2. Test on the second batch of seals

A second batch of seals was supplied in June 2006. A few minor design modifications were made to improve the fit and facilitate the use of the seals.

The seals were read the first time at Ispra with a new reading-head. They were then read again in Cernavoda with a second reading-head. The table below shows the results of these two readings.

| Seal | Correlations |
|---|---|
| JCS001 | 0.980 |
| JCS002 | 0.979 |
| JCS003 | 0.979 |
| JCS004 | 0.981 |
| JCS005 | 0.980 |
| Moy. | **0.980** |

Table 2: Correlations between Ispra and Cernavoda readings (different reading heads)

This result demonstrates that reading heads are interchange. The difference in readings between two reading-heads is clearly inside the acceptable tolerance (correlations of 0.98 compared to the acceptance threshold of 0.93).

### 5.3 Test on the third batch of seals

The second batch of seals behaved as expected, from both the handling and reading points of view. Fifty of these seals were produced to support a Vulnerability Assessment. The IAEA requires a 3$^{rd}$ party vulnerability assessment before a seal can be authorized for safeguards use. A new reading head and inspection software were also produced for this assessment.

All seals were read once with the "old" reading head and then a second time with the new reading head. The correlations between these two set of readings confirmed the results obtained from the second batch of seals. The median of these 50 correlations is 0.983. The measurements were made using the production version of the inspections software

## 6 Conclusion

Ultrasonic seals are based on the same reading method but can be adapted to a wide variety of applications. Their main advantages are that they can handle very high levels of radiation, long life time, and in-situ verification.

JRC Candu Seals can be used in place of existing ARC seals. They use the same tools for handling (installation or removal), only the reading head is different. Reference measurements can be measured and saved at headquarters. This reference measurement will be used for the life of the seal.

Fifty seals and one reading head will be delivered to the IAEA to support both a Vulnerability Assessment and the follow-up for the Cernavoda field tests.

Production and test procedures have been established and the first batch of 50 JCS seals for the vulnerability assessment have been produced.

# Session 9

# NDA-I neutrons and gamma

# Research Reactor Fuel Counter Performance Calculations

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***Abstract:***

*The design of a portable neutron detector for research reactor fuel has been presented recently [1]. The detector measures both passive and active neutron emission and passive gamma emission. The current paper presents the results of the Monte Carlo calculation of its performance for several different fuel types.*

*For the passive mode the neutron emission from spent fuel has been calculated using SCALE. The neutron and gamma emission depends on the enrichment, burnup, cooling time and fuel matrix. The latter has a large effect on the (alpha,n) emission of the fuel. Firstly we consider the case of 5 year cooled fuel with similar percentage burnup. For high enriched oxide fuel, the main neutron emission comes from $^{240}Pu$ and $^{239}Pu$, whereas for low enriched oxide fuel the main neutron emission comes from $^{244}Cm$ and $^{240}Pu$. For a given burnup the neutron emission from the low enriched fuel is greater than the neutron emission from the high enriched fuel whereas the gamma emission is less.*

*The second part of the report analyses real data from a recent measurement [2]. In this case there is a mixture of enrichment, burnup, cooling time and fuel matrix in the measured assemblies. Good agreement is found between the measured neutron and gamma emission and the values calculated using the operator's values. Short cooled LEU fuel can be easily distinguished from longer cooled HEU fuel.*

*The burnup code was then used to predict the neutron and gamma emission from long-cooled LEU fuel. In this case, because the $^{244}Cm$ has significantly decayed, the neutron and gamma emission from the LEU fuel is much more similar to that from HEU fuel. The present calculations have not uncovered a universal analysis method that can distinguish LEU from HEU assemblies under all circumstances. At present it seems that each set of measurements would need to be analysed on a case by case basis.*

**Keywords:** NDA instrumentation, spent fuel, research reactors, neutron, gamma, burnup

## 1. Introduction

The Advanced Experimental Fuel Counter (AEFC) has been designed to measure research reactor fuel in active and passive mode [1]. Figure 1 shows the Monte Carlo model of the detector. In the passive mode the counter measures both the singles count rate and the coincidence count rate from spontaneous fission emission, (alpha,n) emission and induced fission from the fuel. In the active mode there are two measurement possibilities: the net signal on the rear detectors and the coincidence signal on the front detectors are both proportional to the residual fissile material content of the fuel. The counter has been used to make measurements on DIDO type fuel and the results are described in another paper at this conference [2]. This paper describes calculations that have been carried to interpret the results of measurements.

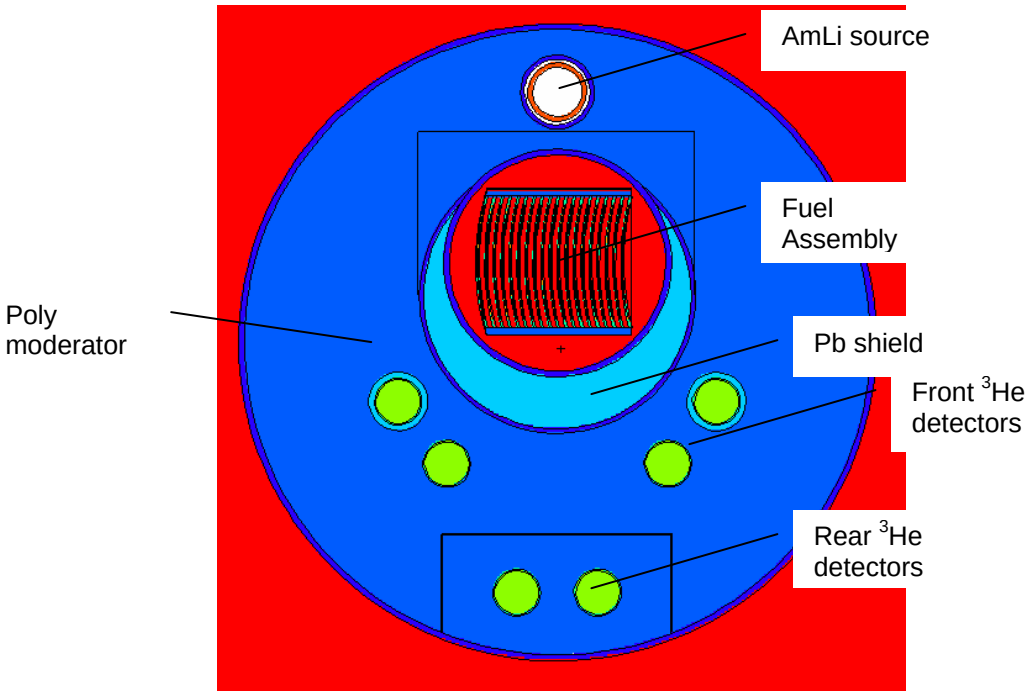


Figure 1 Plot of MCNP model of AEFC

## 2. Passive measurements

The Scale 5.1 [3] package was used to make burnup calculations of DIDO fuel. ORIGEN-ARP cross-section libraries were generated in the standard way for 80%, 60% and 20% enriched fuel. The reactor physics model was made using the TRITON transport code.

### *2.1 Initial calculations*

In addition to different enrichment, the measured assemblies covered a very wide range of important parameters such as burnup, cooling time and matrix type. In order to understand the basic behaviour of different enrichment fuel, the cross section libraries were used to predict the neutron and gamma emission of different enrichment fuel under similar circumstances. The first set of calculations was made for 80%, 60% and 20% enriched fuel with burnup of 14% of the initial $^{235}$U with 5 year cooling. Table 1 shows the passive emission per metric ton of initial heavy metal.

| Enrichment | (alpha,n) n/s | Spontaneous fission n/s | Total n/s | “alpha” | Gamma Emission MeV/s |
|---|---|---|---|---|---|
| 20 | 2.57E+05 | 2.84E+05 | 5.41E+05 | 0.905 | 1.94E+15 |
| 60 | 2.24E+05 | 2.08E+05 | 4.32E+05 | 1.077 | 5.63E+15 |
| 80 | 1.40E+05 | 9.90E+04 | 2.39E+05 | 1.414 | 7.47E+15 |

Table 1 Neutron emission from 5 year cooled oxide fuel

The total neutron emission from the fuel decreases as the enrichment increases. The contributions to the neutron emission from (alpha,n) and from spontaneous fission are of similar size for this oxide fuel, but with a greater contribution from (alpha,n) for the higher enrichment fuel. Figure 2 shows the relative contributions to the neutron emission. The gamma emission is higher for higher enrichment, so that the neutron/gamma ratio could in principle be used to distinguish the initial enrichment of the fuel.

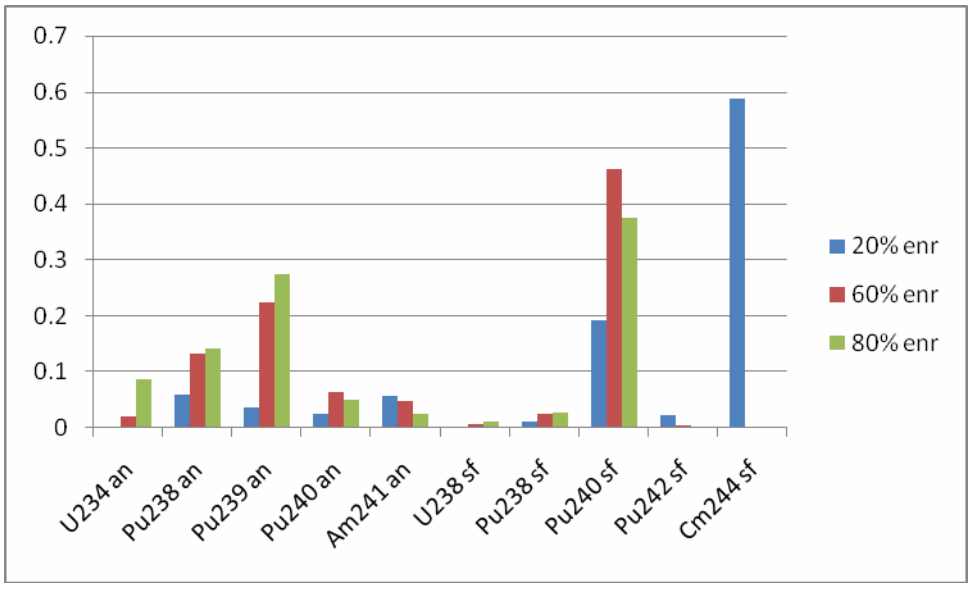


Figure 2 Fractional Contributions to neutron emission from 5 year cooled oxide fuel

It is important to state how the burnup of the fuel is determined. If the different enrichment fuels are in a reactor with the same neutron flux for the same amount of time, the amount of $^{235}U$ consumed will be roughly similar. When this burnup is expressed as Gigawatt-days per metric ton of heavy metal, the values will be inversely proportional to initial enrichment. In this case a 14% burnup corresponds to 22 GWD/MT for 20% enrichment, 64 GWD/MT for 60% enrichment and 85 GWD/MT for 80% enrichment, because the initial amount of $^{235}U$ is similar in each case, but the initial amount of heavy metal is inversely proportional to the enrichment. A calculation for the same fuel type and cooling time but with a fixed burnup of 85 GWD/MT gives total neutron production of 1.23E+07, 7.50E+05 and 2.39E+05 for 20, 60 and 80% enrichment respectively (factors of 50, 3.1 and 1).

The second set of calculations used the same 14% $^{235}U$ burnup and 5 year cooling time, but changed the fuel matrix. ORIGEN-ARP was used to calculate the (alpha,n) emission from a $U_3O_8$-Al matrix and a $U_3Si_2$ matrix. (This calculation is approximate for the $U_3O_8$ case because it assumes uniform mixing of the nuclides.). The results for these cases are plotted together with the results for oxide fuel in Figure 3.

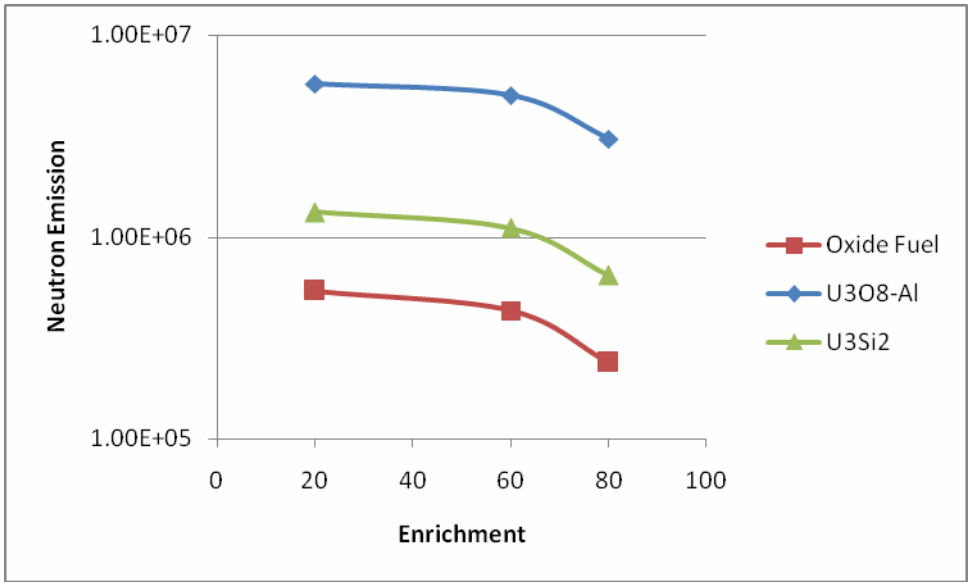


Figure 3 Neutron emission from 5 year cooled oxide, $U_3O_8$-Al and $U_3Si_2$ fuel

The effect of the matrix is a much more important factor than the enrichment in the neutron emission from the fuel. For the $U_3O_8$-Al case the neutron emission is an order of magnitude greater than the oxide fuel. For the $U_3Si_2$ case the emission is around 3 times larger than oxide fuel.

### *2.2 Actual Fuel Calculations*

The same cross-section libraries were used to calculate the neutron and gamma emission from the measured fuel cases in [2].

| Enrichment | Burnup % | Burnup GWD/MT | Cooling time days | Matrix | Neutron emission n/sec | Gamma emission MeV/sec |
|---|---|---|---|---|---|---|
| 20 | 45 | 85.19 | 483 | $U_2Si_3$ | 15386 | 3.98E+13 |
| 80 | 29 | 189.9 | 14849 | $U_3O_8$-Al | 3613 | 1.12E+12 |
| 80 | 40 | 258.0 | 15521 | $U_3O_8$-Al | 7726 | 1.45E+12 |
| 80 | 30 | 195.8 | 16331 | $U_3O_8$-Al | 2722 | 7.88E+11 |
| 60 | 60 | 297.8 | 1435 | $U_3O_8$-Al | 46776 | 1.22E+13 |
| 60 | 51 | 256.5 | 1120 | $U_3O_8$-Al | 29443 | 1.27E+13 |
| 60 | 58 | 317.4 | 1050 | $U_3O_8$-Al | 62761 | 1.90E+13 |

Table 2 Neutron and gamma emission from measured fuel

The measured rates are plotted against the calculated rates for neutrons and gamma in figure 4 and 5 respectively. In both figures the LEU point is marked as a red circle to distinguish it from the HEU blue diamond points. The measurement data agrees very well with the calculations for both neutrons and gammas. This gives confidence in the declared values, the measurements and the calculations.

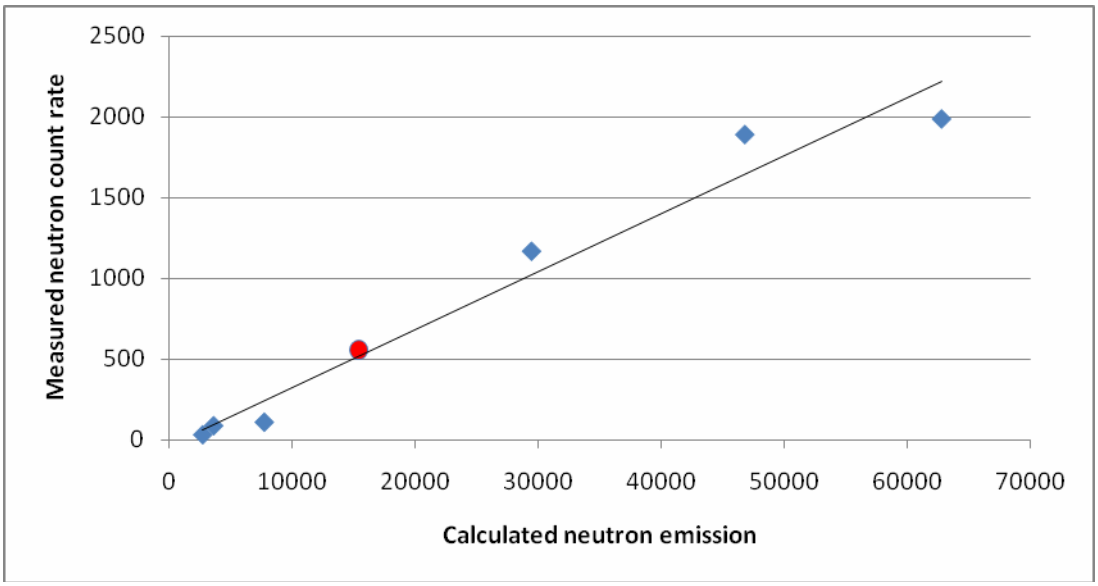


Figure 4 Measured neutron count rate versus calculated neutron emission rate

These results show that the neutron to gamma ratio for these fuel elements does give a clear distinction between LEU and HEU fuel as described in [2].

The relative contributions to this neutron emission are shown in Figure 5. Because of the aluminium in the fuel matrix, the (alpha,n) emission from the HEU fuel is much greater than the spontaneous fission emission.

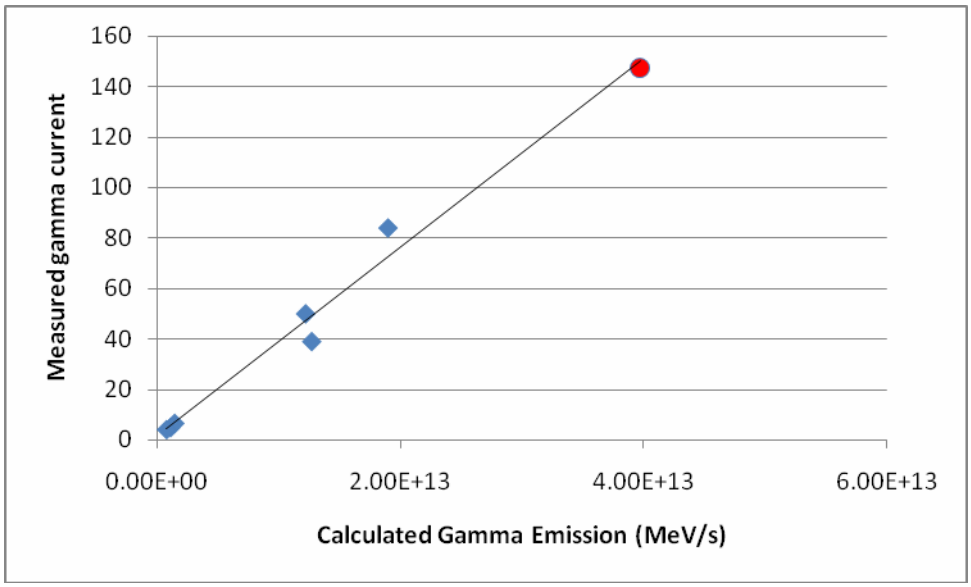


Figure 5 Measured gamma current versus calculated gamma emission rate

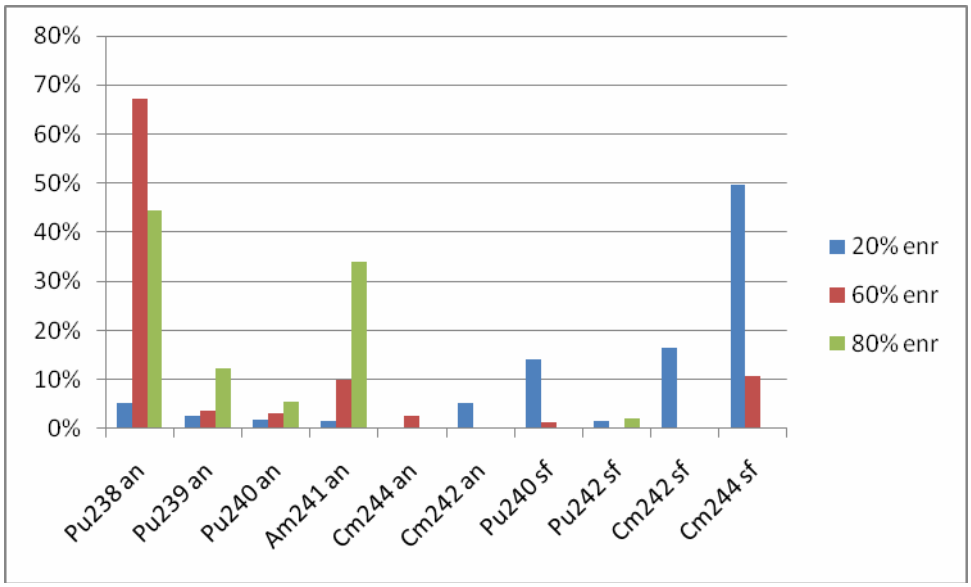


Figure 6 Fractional Contributions to neutron emission from measured fuel assemblies

### *2.2 Decay Calculations*

If we decay the LEU assembly to 1000 and 16000 days in order to be more like the decay times of the 60% and 80% enrichment fuel respectively, the neutron to gamma ratio moves towards the long cooled 80% enrichment case. The results are shown in figure 7. As the LEU assembly cools, the neutron emission falls more slowly than the gamma emission as increased (alpha,n) emission replaces the reduction of spontaneous fission emission from $^{244}$Cm. This behaviour seems to imply that it would not be possible to distinguish between long-cooled LEU and long-cooled HEU fuel by the use of the neutron/gamma ratio.

Figure 8 shows the contribution to the neutron emission from different nuclide for 100 day cooled and 16000 day cooled LEU fuel. The ratio of (alpha,n) emission to spontaneous fission (“alpha”) is around 0.2 until about 100 days, but the decay of $^{244}$Cm and the increase in $^{241}$Am leads to a ratio of 1 at 16000 days. A comparison of the (alpha,n) to spontaneous fission rate for various cases is shown in Table 3.

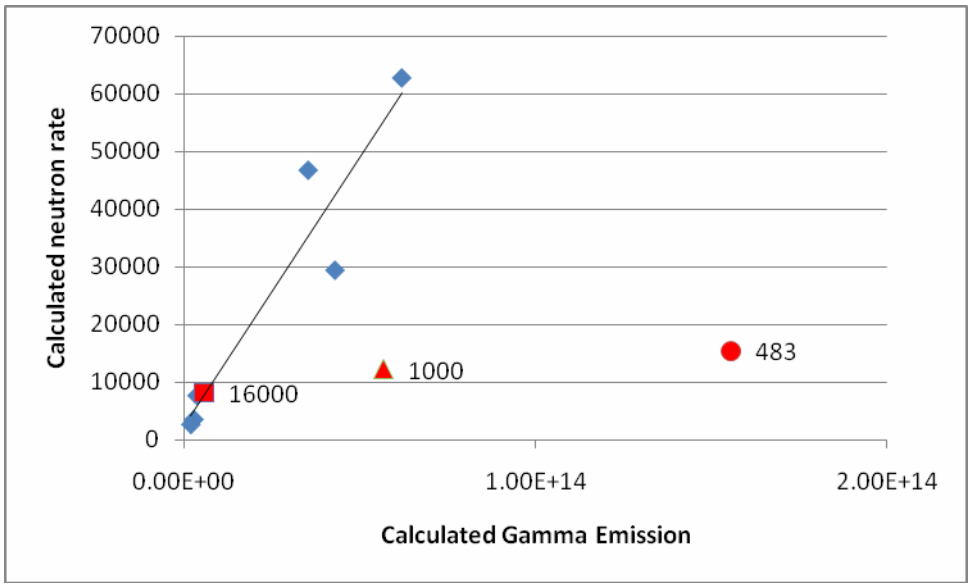


Figure 7 Calculated neutron rate versus calculated gamma emission rate. The HEU points are blue diamonds and the LEU point is a red circle for 483 days cooling, a red triangle for 1000 days cooling and a red square for 16000 days cooling.

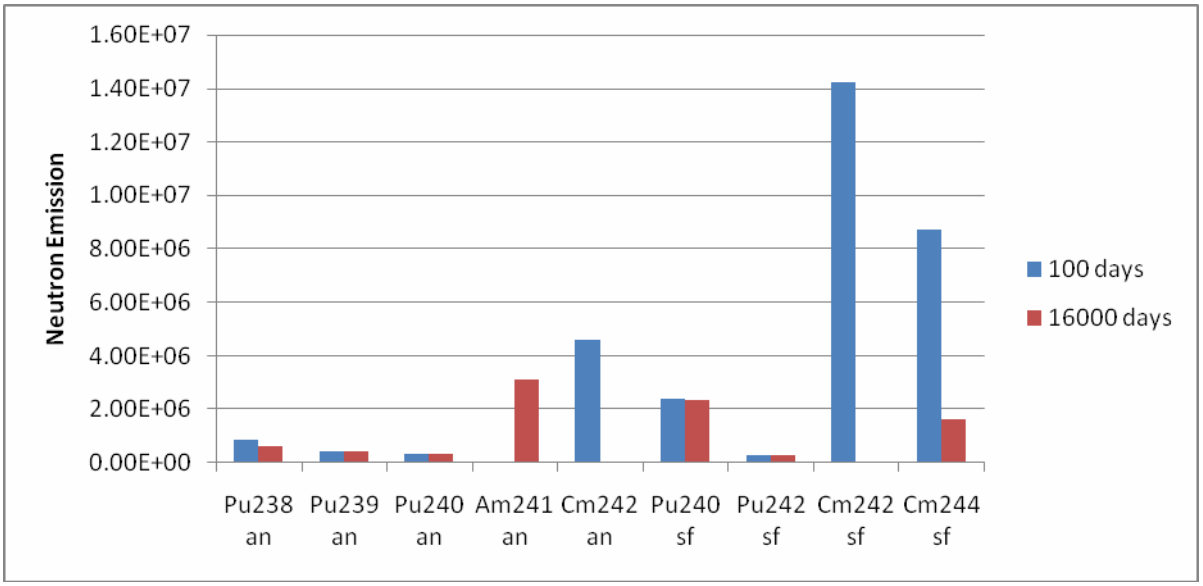


Figure 8 Contributions to the neutron emission for different cooling times for an LEU assembly.

| Case | Cooling time | Matrix | alpha |
|---|---|---|---|
| LEU | 483 | $U_3Si_2$ | 0.21 |
| LEU | 1000 | $U_3Si_2$ | 0.21 |
| LEU | 16000 | $U_3Si_2$ | 1.03 |
| LEU | 1000 | $U_3O_8$-Al | 4.99 |
| 80% | 16000 | $U_3O_8$-Al | 40 |
| 60% | 1000 | $U_3O_8$-Al | 5-10 |

Table 3 (alpha,n) to spontaneous fission rates for different fuel matrices and cooling times.

The alpha value is important because it affects the ratio of neutron coincidences to neutron singles rate as described in [2].The results in Table 3 show that for the experimental case the alpha value of

the LEU fuel was 0.21 compared to 5-10 and 40 for the HEU cases. If the cooling time of the LEU fuel had been very long, this alpha value would have increased to 1.0, making it more difficult to distinguish from the LEU. Similarly if the LEU matrix had been $U_3O_8$-Al the alpha value would have been similar to one of the HEU cases. In the general case it could be difficult to determine the initial enrichment of the fuel by relying on the alpha ratio.

## 3. Conclusions

The calculations of the passive neutron emission from DIDO fuels agree well with expected values. In the case of the current measurements [2] the parameters of the fuel lead to neutron count-rates (singles and coincidences) and gamma count rates that allowed HEU to be distinguished from LEU assemblies. The present calculations have not uncovered a universal analysis method that can distinguish LEU from HEU assemblies under all circumstances. At present it seems that each set of measurements would need to be treated on a case by case basis.

## 4. References

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# New trends in neutron coincidence counting: digital signal processing

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***Abstract:***

*Neutron coincidence counting is the reference NDA technique used in nuclear safeguards to measure the fissile mass in nuclear material samples. Nowadays most of the neutron counting systems are based on the original shift register technology, like the (ordinary or Multiplicity) Shift Register Analyser. The analogue signal from the He-3 tubes is processed by amplifier/SCA producing a train of logical TTL pulses that are fed into a neutron analyser performing the time correlation analysis. In the future these systems could be replaced by high-speed PC's equipped with pulse acquisition cards, providing a time stamp (LIST mode acquisition) for every digital pulse. The time stamp data can be processed directly during acquisition or saved on a hard disk. The latter method has the advantage that measurement parameters, like for instance the pre-delay and gate-width, can be modified without repeating the acquisition. The use of PC based instruments, also called virtual instruments, could be the future major development in practical neutron correlation analysis.*
*Under a push from the main inspection authorities (IAEA, Euratom and French Ministry of Industry) several research laboratories have started to study and develop prototypes of neutron counting systems with PC-based processing. A collaboration in this field among JRC, IRSN and LANL has been established within the framework of the ESARDA-NDA working group. Joint testing campaigns have been performed in the JRC PERLA laboratory, using different equipment provided by the three partners.*
*The paper will describe the rationale for changing to the new technology, give an overview of the hardware and software tools available today and a feedback of the experience gained in the first tests. Associated to the experimental tests, the ESARDA-NDA working group is also performing an inter-comparison benchmark exercise on the analysis software for pulse processing.*

## 1. Introduction

Since more than thirty years the measurement of the mass of special nuclear materials is done in non-destructive way using neutron coincidence counting or, more recently, neutron multiplicity counting. This technique relies on the concept that in plutonium the mass is proportional to the spontaneous fission rate (passive NCC) and in uranium the fissile 235U mass is proportional to the induced fission rate (active NCC). Since neutron are also generated by other competitive phenomena, such as background from cosmic and surrounding materials or (alpha,n) reactions, the discrimination of fission (producing several correlated neutrons) from other reactions (producing single uncorrelated neutrons) is based on the capability to analyse the time correlation between neutron events.

The time correlation analysis is traditionally performed through a dedicated electronics. The analogue signal from the He-3 tubes is processed by amplifier/SCA producing a train of logical TTL pulses that are fed into a neutron analyser based on the shift-register principle. This kind of equipment has worked remarkably well through at least three decades, but starts to show the first symptoms of obsolescence. One of the major limitations of current shift-registers are the incapability to deal with very high count rates, but other features would be required in order to improve the neutron counting technique, such as the possibility to take into account in the analysis the position of the detected event.

For these reasons the IAEA and Euratom have expressed the need to investigate new avenues for a novel concept of neutron electronics that would override the weaknesses of the shift-register technology. The continuous improvement in computer science, both in terms of computing speed and in mass storage capabilities, has induced a recent trend in many experimental applications (not limited to nuclear physics). Instead of processing signals through analogue electronic modules, the current tendency is to acquire and store large quantities of raw digital information and then process it via software.

The ESARDA NDA working group, a forum collecting worldwide specialists on NDA measurements applied in safeguards, has accepted the challenge and stimulated projects with the aim to investigate, test and eventually implement some alternatives for a digital signal processing of neutron coincidence counters. This addresses the developments of the two major components: the digital electronics and the processing software.

In this paper we will present some recent results produced within these projects. In the field of electronics three laboratories have compared the performances of three different digital acquisition system in two laboratory campaigns performed in the PERLA laboratory at Ispra. In the field of software, the NDA working group has launched two benchmarks for comparing the different software to analyse LIST mode pulse trains.

## 2. New concepts for digital signal processing

There are two different approaches currently under study and both of them have been analysed in the PERLA campaign under identical conditions (using the same samples): the LIST mode and the virtual instrument methods.

### 2.1. LIST mode acquisition and processing

The LIST mode concept is based on a time stamping technique. After each detection event the time at which the detection happened is sent to the PC (more often the datation cards provide the time elapsed between two successive events, but this can be easily converted to an absolute time sequence). Optionally the information of the channel where the detection happened can be added in order to use the location information when needed. The results of the acquisition is a (possibly large) pulse train file with all the information for each individual detection event (time, channel, others).

The pulse train can be then analysed off-line with a processing software that simulates the operation of a shift register (pre-delay, opening of the R+A gate, scaler counting, delayed gate for accidentals, storage of multiplicity distributions) and to compute all the physical parameters needed: Singles, Doubles, Triples and associated uncertainty.

### 2.2. The virtual instrument approach

The virtual instrument is a neutron analyser integrated in the neutron counter. It is basically a digital card which reproduces directly all the functionalities of the shift-register. It directly integrates the counts within the different gates and transmits directly the Singles, Doubles and Triples to the acquisition PC.

The virtual instrument approach presents advantages and disadvantages with respect to the LIST mode. The major disadvantage is that the shift-register characteristics must be preset (like in a real instrument), whereas in the LIST mode the operational parameter are simulated by the software with the consequence that the same pulse train can be reanalysed using different instrumental setups.

On the other hand the main advantage is that the data transmitted to the PC are already the integrated counts in each time step, meaning that the computing effort is minimal requiring only an averaging of the values, so the results are provided immediately at the end of the acquisition. The LIST mode requires an off-line post-processing of the pulse trains introducing a delay in the availability of the result. In case of high number of events (long acquisitions or high count rates) the dimension of the pulse file can be huge and the processing time increases exponentially (in some cases up to several hours).

## 3. Experiments with new electronics for neutron multiplicity counting

During two experimental campaigns in the PERLA laboratory at JRC, three different configurations prepared by three different institutions (IRSN, LANL and JRC) have been tested. A large number of measurements have been done, but the comparison focuses on six of them that were repeated with the three setups on the same samples:

- a low intensity californium source
- a high intensity californium source
- a small (20 g) Pu metal sample
- a small (50 g) Pu oxide sample
- a large (1000 g) Pu oxide sample
- a large MOX sample

The three laboratories used different counters and different digital electronics, but in all cases the measurements were repeated with the novel digital system and with a traditional shift-register analyser in order to compare the results and validate the performances of the novel electronics.

### 3.1. The MEDAS card and the TRIDEN software developed by CESIGMA and IRSN

The IRSN setup consisted in an AWCC (Active Well Coincidence Counter) operated in passive mode and in fast configuration with Cd liners. All the acquisitions were repeated using a conventional neutron analyser (a CANBERRA 2150) and the so-called MEDAS datation card connected to the TTL output of the AWCC.

The system MEDAS (Multi Event Datation System) is an electronic card, designed by the company CESIGMA to acquire logical signals coming from any measurement system in several modes: counting, multi scale counting or time stamping mode [1]. The system is able to date and count physical events of input frequency up to 10 MHz simultaneously on 32 inputs with a maximum accuracy of 25 ns. The capacity of recording depends then on the memory available. Typically, counts rate about 150000 counts per second creates a 100 Mb binary file for 100s measurement time. The MEDAS output binary file is converted to ASCII for further processing.

A software, named TRIDEN (Treatment of Dated pulses coming from Neutron Emitters) has been developed by IRSN to process the digital pulse train issued from MEDAS and compute count rates: singles (S), doubles (D) and triples (T) taking account the correct setup of detection cell (gate width, predelay and long delay). This software basically consists in:

- simulating the operation of a multiplicity shift register starting from the first pulse detected up to the one detected some milliseconds before the end of the measurement, each one opens, after a predelay, a first time gate to collect the number of real+accidental (R+A) pulses detected and, following a long delay, a second time gate to collect the number of accidentals pulses (A);

- calculating the neutron count rates of interest: S, D and T from the multiplicity distributions in the “R+A” and “A” gates (TRIDEN offers also the possibility to split an initial binary file in several parts in witch it calculates S, D and T and the associated standard deviation);

- deriving from the point model equations [2], knowing the isotopic composition, the mass of plutonium resulting from a basic passive neutron counting measurement.

During the experimental campaigns at PERLA laboratory, the same procedure was applied for all the sources or samples measured with the AWCC : 10 acquisitions of 100 seconds each were performed successively with both systems: the AWCC connected to a 2150 electronics driven by the INCC software, then the AWCC connected to the MEDAS system, using TRIDEN software with the same parameters than the ones integrated in INCC (predelay of 4,5 μs, gate width of 64 μs, long delay of 1024 μs, no dead-time corrections) to analyse the resulting pulse trains.

Table 1 shows a comparison between the singles (S), doubles (D) and triples (T) estimated using the two following systems: AWCC+2150+INCC and AWCC+MEDAS+TRIDEN, for each of the 6 samples introduced above. For each count rate, the table gives first the relative discrepancy between MEDAS and 2150 (taking 2150 as reference), then the relative uncertainty given by TRIDEN (standard

deviation observed on the distribution of the count rates on 10 acquisitions of 100 s), and finally the relative uncertainty given by INCC (issued from theoretical equations [2]).

| | Comparison MEDAS / 2150 | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| Reference | ΔS | σS Triden | σS Incc | ΔD | σD Triden | σD Incc | ΔT | σT Triden | σT Incc |
| 6005NC | -0.1% | 0.0% | 0.1% | 0.2% | 0.2% | 0.2% | 0.1% | 0.6% | 0.7% |
| 6001NC | 0.0% | 0.0% | 0.0% | 0.1% | 0.2% | 0.2% | -11.4% | 4.4% | 4.0% |
| 211 | 0.1% | 0.0% | 0.2% | 1.7% | 0.4% | 0.4% | 0.1% | 2.1% | 1.2% |
| 102 | 0.0% | 0.0% | 0.1% | -0.5% | 0.4% | 0.4% | -1.9% | 1.9% | 2.6% |
| 111 | 0.0% | 0.0% | 0.0% | 0.2% | 0.3% | 0.2% | -3.8% | 7.1% | 7.0% |
| ENEA01 | 0.0% | 0.0% | 0.0% | -0.4% | 0.3% | 0.5% | -8.3% | 3.9% | 6.4% |

Table 1 - Comparison : S, D, T issued from MEDAS vs 2150 with AWCC

The S, D and T rates coming from MEDAS are very consistent with the ones issued from the 2150 electronics presented here without any dead-time corrections. The uncertainties calculated from the repetition of the measurement are quite small especially for the single and the doubles, indicating a good reproducibility of the count rates. The relative discrepancy observed on S and D rates, while taking as reference the 2150 electronics, is lower than 1%. For the triples rates, the counting statistics are lower and this directly impacts the scattering around the mean value of the 10 counting cycles. The discrepancy on the triples is higher than the one observed on S and D rates but remains lower than the counting uncertainty in the majority of the cases except for samples having a high neutron emission rate. One explanation to this discrepancy is to consider that the predelay applied to the MEDAS card is not optimized. Indeed, if the predelay adjustments are located downstream from the Rossi-Alpha curve, the detection probability between two or more consecutive events is not maximum, which is not important for low intensity sources, but which has a significant impact for high intensity sources (6001NC, 111 and ENEA01).

### 3.2. The LMMM data acquisition system developed at LANL

The LANL setup consisted of an ENMC (Epithermal Neutron Multiplicity Counter) [3]. The detection efficiency is around 64%. The ENMC was modified to provide both a conventional output to an AMSR and a differential signal each of the 27 Amptek A11 preamplifier/discriminators.

The AMSR was used in parallel with a list mode data acquisition system developed by LANL. It consists of a unit (LMMM = List Mode Multiplicity Module) that receives the 27 differential pulses and records the arrival time of each. The List Mode Counter core consists of pulse detection logic for up to 64 external sensor inputs, a buffering FIFO, and a timestamp unit, and is written in VHDL. This core is loaded into a Field Programmable Gate Array (FPGA) chip and interfaces to a Power PC (PPC405) core embedded into the FPGA. The PPC runs an embedded microkernel at 400 MHz which collect the list mode data into blocks that are sent to the Windows PC via Ethernet via UDP socket protocol. The bins size or time resolution cans be set as low as 10 ns, but is normally set at 100ns.

An application running on the Windows PC receives and stores the packets in binary files. This application is used to control the acquisition time. During these experiments the data acquisition was run continuously with 100 second acquisition time. The data acquisition rate is currently limited to 100 Khz. This limitation is caused by the code that loads the data onto the Ethernet.

The resulting binary files are read by a VisualBasic program that calculates Singles and Doubles and can optionally create an ASCII pulse file. These pulse files are identical in format to the pulse files used in the ESARDA multiplicity benchmark exercise [4] and can therefore be read with the software used in that exercise. This is done when Triples rates are required.

Table 2 shows the results from the ENMC for the samples considered above. The numbers in the table are not deadtime or background corrected. A comparison between the results shows that the agreement in the Singles is 0.06% or better. The list mode Doubles are systematically 0.5% lower than the AMSR doubles. This is caused by the digital synchronization of the pulses. In the AMSR the clock interval is 0.25μs, which effectively increases the gate by up to this amount – roughly half of the interval. In the LMMM the clock interval is 0.1μsec, so the relative gate lengths are 24.125/24.05

=1.003. The effect is similar in principle for the Triples where we see approximately a 1% difference between the AMSR and LMMM results.

| | S (AMSR) | error | S (LMMM) | error | Delta S |
|---|---|---|---|---|---|
| **Small Cf** | 2525 | 0.9 | 2525 | 1 | **0.000%** |
| **Large Cf** (i) | 305434 | 27 | | | |
| **Small Pu metal** | 1536 | 1.5 | 1536 | 1.7 | **0.000%** |
| **Small Pu oxide** | 14861 | 4.5 | 14852 | 4.6 | **0.061%** |
| **Large Pu oxide** (i) | 299526 | 16 | | | |
| **Large MOX** (ii) | 57136 | 6.4 | 57155 | 17 | **-0.033%** |

| | D (AMSR) | error | D (LMMM) | error | Delta D |
|---|---|---|---|---|---|
| **Small Cf** | 1465 | 0.7 | 1472 | 0.9 | **-0.478%** |
| **Large Cf** (i) | 180276 | 118 | | | |
| **Small Pu metal** | 504.5 | 1.1 | 507.3 | 1.2 | **-0.555%** |
| **Small Pu oxide** | 3475 | 3.5 | 3483 | 3.7 | **-0.230%** |
| **Large Pu oxide** (i) | 91587 | 62 | | | |
| **Large MOX** (ii) | 12438 | 11 | 12482 | 41 | **-0.354%** |

| | T (AMSR) | error | T (LMMM) | error | Delta T |
|---|---|---|---|---|---|
| **Small Cf** | 504 | 0.6 | 508.8 | 0.8 | **-0.952%** |
| **Large Cf** (i) | 53168 | 211 | | | |
| **Small Pu metal** | 110.5 | 0.9 | 111.6 | 0.9 | **-0.995%** |
| **Small Pu oxide** | 831.4 | 2.6 | 833.7 | 3.4 | **-0.277%** |
| **Large Pu oxide** (i) | 32670 | 166 | | | |
| **Large MOX** (ii) | 2556 | 11 | 2625 | 52 | **-2.700%** |

(i) Note that the count rate was too large to transfer the data to the Windows PC.
(ii) The AMSR measurement time was 2000 seconds and the LMMM measurement time was 285 seconds, which is why the uncertainties are larger.

Table 2a, 2b, 2c - AMSR and list mode results for the ENMC

### 3.3. The MI-PTA card used at JRC

The JRC setup consisted in a SNMC (Scrap Neutron Multiplicity Counter) [5]. The digital acquisition in the SNMC is not a LIST mode, but it integrates a virtual instrument electronics, the MI-PTA card [6].

The electronic system, called Multi Input-Pulse Train Analysis (MI-PTA), can be used with standard instruments such as the Shift Register or replace it with a direct process of the digital pulse train by PC. The MI-PTA has been designed to perform a variety of pulse train analysis independently or combined with a PC to which the system is connected via a high speed USB2.0 connection. This system has been developed with a maximum of 128 input channels, which can handle count rates in excess of one million counts per second without count losses. Pulse data can be exchanged between the MI-PTA and PC at the rate of 480 Mb/s allowing real time analysis on the PC. The system consists in a number of base units each with 8 TTL compatible inputs. Each input channel has a dedicated 3-bit counter, which is much faster than the input pulse frequency. Counter states are fed through double stage flip-flops to avoid meta-stability. The DSP compares actual counter states with previous states to calculate the number of pulses that arrived during each acquisition cycle (1 μs). One unit, configured as a master, is connected to the PC. As many as 15 slave units can be connected to the master via a High Speed Serial Bus (in the SNMC a master and 5 slaves are needed to manage the 44 amplifiers). An additional feature of the system is that it is soft-configurable: a program running on the PC can monitor automatically a single input at the time. This is very useful for to control of the optimum

working conditions of counter tubes and amplifiers. For instance a failure of a single tube can be detected easily even in a system with more than 100 counter tubes, as it is the case with the SNMC.

The SNMC has also a standard BNC output for TTL signal, so it can also be used with traditional neutron analysers, such as AMSR or JSR-14. Table 3 shows the comparisons of the measurements of the 6 samples with the SNMC using acquisitions by the digital MI-PTA output and PC software analysis with respect to shift-register analyser (JSR-14).

| | S (JSR-14) | error | S (MI-PTA) | error | Delta S |
|---|---|---|---|---|---|
| **Small Cf** | 2377 | 2 | 2377 | 3 | **0.026%** |
| **Large Cf** | 290952 | 24 | 290944 | 24 | **0.003%** |
| **Small Pu metal** | 1502 | 2 | 1503 | 2 | **-0.019%** |
| **Small Pu oxide** | 14413 | 6 | 14410 | 5 | **0.018%** |
| **Large Pu oxide** | 290081 | 18 | 290072 | 19 | **0.003%** |
| **Large MOX** | 55999 | 7 | 56004 | 8 | **-0.009%** |

| | D (JSR-14) | error | D (MI-PTA) | error | Delta D |
|---|---|---|---|---|---|
| **Small Cf** | 1295 | 3 | 1292 | 3 | **0.202%** |
| **Large Cf** | 156024 | 135 | 155225 | 179 | **0.514%** |
| **Small Pu metal** | 451 | 2 | 450 | 2 | **0.188%** |
| **Small Pu oxide** | 3062 | 5 | 3066 | 7 | **-0.150%** |
| **Large Pu oxide** | 79180 | 167 | 78721 | 160 | **0.583%** |
| **Large MOX** | 11089 | 20 | 11032 | 19 | **0.520%** |

| | T (JSR-14) | error | T (MI-PTA) | error | Delta T |
|---|---|---|---|---|---|
| **Small Cf** | 415 | 3 | 415 | 3 | **0.159%** |
| **Large Cf** | 39826 | 503 | 39813 | 472 | **0.033%** |
| **Small Pu metal** | 96 | 5 | 95 | 5 | **0.424%** |
| **Small Pu oxide** | 686 | 7 | 687 | 6 | **-0.155%** |
| **Large Pu oxide** | 25353 | 604 | 25445 | 360 | **-0.361%** |
| **Large MOX** | 2117 | 32 | 2118 | 22 | **-0.034%** |

Table 3a, 3b,3c – Comparison digital vs analogue with JRC setup

## 3. Benchmarks on the processing software

In 2003 the ESARDA NDA working group has launched a benchmark in order to compare the different algorithms and codes used in the simulation of neutron multiplicity counters. In order to derive the maximum of information and at the same time to allow a large participation, the working group decided to split the exercise in two parts with two participation levels: a full simulation exercise where participants were asked to compute the count rates starting from the basic technical specifications and/or a partial exercise involving the processing of the pulse trains produced by a single laboratory. The results of participants performing the entire exercise allowed to make a comparison among the different Monte Carlo codes for the simulation of neutron multiplicity counters. The results of the partial exercise help to test the available algorithms for pulse train analysis and to derive some important information about the models applied for dead-time correction. The results of the 1$^{st}$ and 2$^{nd}$ phase of the ESARDA Multiplicity Benchmark have been published in a special issue of the ESARDA Bulletin [4] and a summary presented and published in the proceedings of the 47$^{th}$ INMM Annual Meeting [7].

All the cases run in the first two phases of the benchmark were totally theoretical. So the conclusions derived had to be considered as a relative behaviour of the different models, techniques and codes.

Notwithstanding the satisfactory conclusion that all the algorithms developed by the different participants in the first two phases and used to analyse the pulse trains have proven to be satisfactory, the working group felt that an extension to real experimental cases would have added a supplementary value to the exercise.

Therefore it was decided to use the experimental campaign described in section 3 for a continuation of the 2003 benchmark. Again it will contain a first step (3rd phase) devoted to full Monte Carlo simulation and a second one (4th phase) for the inter-comparison of software for analysis of LIST mode files. For this benchmark we selected the 6 measurements performed with the IRSN set-up (AWCC with MEDAS card). The LIST mode acquisitions (for each case 10 repeated measurements of 100 s each) have been stored in binary pulse train files and distributed to the participants who will test their software computing Singles, Doubles and Triples rates. These will be compared to the reference: the S, D and T rates obtained with an analogue-electronic acquisition (AMSR).

## 4. Conclusions

One of the interests of pulses datation is to be able to process the signal by changing one or several measurement parameters (predelay, gate width, etc). With a classical mode of acquisition based on a shift register, it will be much more difficult to obtain comparable information about the quality of the result. With an acquisition mode based on dated events, it becomes possible in a short time with a post-processing treatment to determine and optimize characteristics of the measurement cell (die away time, gate width, etc). By calculating the neutron die-away time in the system, one can ensure the quality of a result and confirm that the product is not too polluted by (alpha, n) reactions.

The results presented in this paper show that digital acquisition is rapidly approaching maturity. All the systems tested perform in a way totally comparable to shift register acquisition at low and medium count rates. Some systems present limitations at very high count rates, but this can easily be improved in the near future.

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# Improving the Accuracy of a Uranium Enrichment Monitor Based on a NaI(Tl) Spectrometer and Transmission Source

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***Abstract:***

An enrichment monitor based on a NaI(Tl) spectrometer and a transmission source is very sensitive to any error in the intensity of the transmission peak used to determine the density of $UF_6$ gas. This is because the attenuation of the transmission peak is only a few percent and the measurement scales as the natural log of a transmission fraction near unity. At 97% transmission, an error as small as 0.1% in the attenuation of the transmission-peak count rate will result in a 3% error in the enrichment calculation. The long-term stability of the transmission-peak count rate is also important because the enrichment formula uses an initial value of that count rate through an empty pipe as a constant during the time between two calibrations. The demand for such an unusually high level of accuracy challenges the capabilities of off-the-shelf NaI(Tl) spectrometers, especially at the low pressures of the $UF_6$ gas used in fuel-processing facilities.

The effects of environmental factors, such as temperature, pickup noise, and geometry stability, on the measured enrichment have been evaluated. We have found that these environmental factors affect the count-rate information by interfering with and creating error in the dead-time correction. Even though the gain stabilization system holds the transmission peak at a specified energy channel, a variation of +/–10 $^{o}$C in the room temperature can cause more than a 10% change in the measured enrichment. High-frequency pickup noise affects the dead-time correction system and, hence, the measured count rates. This can occur without any observable changes in the pulse-height spectrum.

By using a patented technology in the NaI(Tl) detector preamplifier, we are able to reduce the temperature error by more than an order of magnitude and improve the count-rate characteristics of the NaI(Tl) spectrometer. A simple method for diagnosing the noise immunity of already installed instrumentation will be presented along with our experience in reducing noise sensitivity in harsh industrial environments.

Our efforts and the experimental results for further reduction of these errors, by using modified hardware for enrichment monitoring, will be outlined.

**Keywords:** uranium enrichment monitor, measurement accuracy, uranium hexafluoride, sodium iodide detector

## 1. Introduction

Our instrumentation consists of a single transmission source and a NaI(Tl) detector with a multichannel analyzer (MCA) to measure the enrichment of $UF_6$ gas [1-3]. This system measures both the attenuation of the intensity of the transmission source peak ($^{241}$Am, 60 keV, or $^{57}$Co, 122 keV) by the $UF_6$ gas and the intensity of the 186-keV gamma ray from $^{235}$U. The enrichment is calculated by the formula

$$E = K \times (R - B)/\ln [I/I_0(t)] \quad ,$$

where

$E$ = enrichment (%) of $^{235}$U in the $UF_6$ gas,
$K$ = calibration constant,
$R$ = count rate of $^{235}$U, 186 keV, from $UF_6$ + background,
$B$ = count rate of $^{235}$U, 186 keV, from background,
$I_0(t)$ = count rate of the transmission source peak without $UF_6$ gas in the pipe, corrected for the source decay,
$I$ = count rate of the transmission source peak with $UF_6$ gas in the process pipe, and
$I/I_o$ = the transmission fraction of the transmission source peak through the process pipe and $UF_6$ gas.

The dead time (DT) that results from the microseconds-long NaI(Tl) light pulses and the MCA shaping time causes losses in the count rates of the peaks. Because the attenuation of the transmission peak used to measure the density of $UF_6$ gas is only a couple of percent and that attenuation is used for the full scale of the enrichment, the errors in the transmission-peak ratio propagate through the enrichment calculation and can be multiplied by a factor of 30 or more in some situations. For instance, a 0.1% error in the transmission ratio $I/I_0$ leads to an error of a couple of percent in the enrichment. The data in Table 1 show some examples of the effect of the multiplication factor for different combinations of source energy, pipe material, and gas pressure. Note that the intensity $I_0$ of the transmission source peak without $UF_6$ gas is measured only once during the calibration; therefore, the total instability of the counting information should be kept below 0.1% over a period of many months. Our observations on already installed equipment show long-term drift in the counting information, even though the MCA stabilization system keeps the transmission-peak position very stable. This level of accuracy is in the range of high-resolution high-purity germanium (HPGe) spectrometers. Unfortunately, the use of high-resolution detectors that require cryogenic cooling, such as HPGe spectrometers, is not possible for this application, and cost is an issue. Because of requirements for good attenuation in the $UF_6$ gas, low-energy transmission sources such as $^{57}$Co, $^{241}$Am, and even $^{109}$Cd are the preferred choices. The feasible detection alternatives for that energy range and a harsh industrial environment are the scintillation detectors—especially the radiation industry workhorse, the NaI(Tl) scintillation detector. These detectors are famous for the complexity of their light response and the difficulties of stabilizing the energy calibration of their pulse-height spectrum. The stabilization of the counting information is much more complex and, generally, is beyond the capabilities of the available off-the-shelf instrumentation. One unattended enrichment monitoring system based on that principle is the CEMO (continuous enrichment monitor) system [4]. In order to overcome the above instabilities, this system uses the 22-keV line from $^{109}$Cd (which is strongly attenuated by the $UF_6$ gas) for the transmission line and the 88-keV high-energy line from $^{109}$Cd for spectrum stabilization. Use of such a low-energy transmission line requires a special $\leq$5-mm-thick aluminum window on the $UF_6$ pipe. The use of such a special window is not convenient and, in many cases, is not an option. Using the 88-keV line for gain stabilization does not guarantee a stable peak position for the 22-keV line. The nonlinearity of the light output, especially its temperature dependence, can cause temperature drift of the 22-keV peak even though the 88-kev peak position is stable [5]. The changing intensity and spectrum shape due to transmission-source decay presents another set of challenges related to the total count rate and calculation of the intensities of the transmission and uranium lines.

| Source | Energy (keV) | Pipe ID* (mm) | Pipe Material | Pipe Wall Thickness (mm) | Gas Pressure (torr) | Transmission Fraction | Multiplication Factor |
|---|---|---|---|---|---|---|---|
| $^{241}$Am | 60 | 97 | Al | 6 | 45 | 0.96 | 24 |
| $^{241}$Am | 60 | 203 | Al | 8 | 51 | 0.92 | 12 |
| $^{57}$Co | 122 | 100 | Fe | 4 | 38 | 0.97 | 33 |

* ID = Inside diameter

**Table 1:** Representative measured pipe data.

In this work, we aim to refine the accuracy of the classical enrichment monitor (EM) and facilitate its application in enrichment measurements; for example, by the use of a higher energy source or an extended maintenance period. The main factors that affect the long-term stability are temperature, noise, and systematic changes to the intensity and spectrum shape due to a decaying source. We analyze the sensitivity to the above factors and describe our efforts to build a NaI(Tl) spectrometer that can address the high-accuracy requirements. At the end of the paper, we address other causes of sytematic errors that result from interference between the spectral lines of the transmission source and $^{235}$U lines, as well as the affect of the changing spectrum shape on the calculation of the transmission-peak net area.

## 2. Temperature sensitivity of the enrichment results

Because of the overwhelming effect of the transmission ratio $I/I_o$ on the enrichment, we focused on the temperature dependence of the transmission peak. In our previous work [5–7], we described the temperature behavior of NaI(Tl) scintillators as a linear dependence of the total light output and a temperature-dependent redistribution of two light components (fast and slow). The temperature affects the pulse-height-analysis parameters (peak position, energy resolution, and nonlinearity) by way of a convolution between the temperature-dependent shape of the light pulse and the finite time response of the shaper. The profound effect of that convolution can be seen in Fig. 1, where the temperature drift is shown for different shapers.

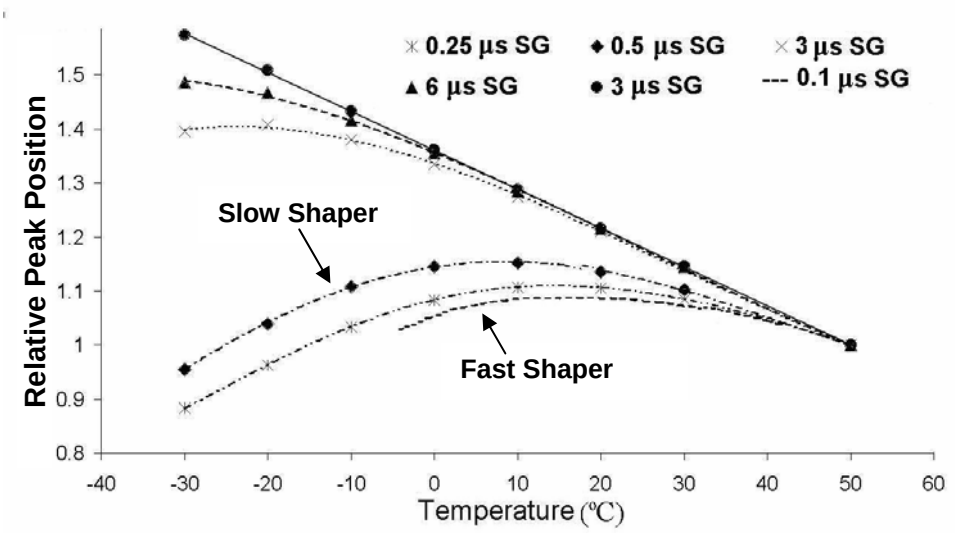


**Fig. 1***:* Temperature dependence of peak position and FWHM for various time constants with semi-Gaussian (SG) shapers in the MCA [6]. The characteristics for the 0.1-µs shaper are included to illustrate the difference between the temperature behavior of the fast and slow channels.

Most modern MCAs have a digital gain stabilization system that practically eliminates the temperature effect on the position of the stabilization peak. However, the temperature dependence will cause a drift in the positions of other peaks in the spectrum because the nonlinearity of the light output is temperature dependent [7]. The stabilization of the counting information is much more complicated

because it is affected by many factors, such as the full width at half maximum (FWHM) of the transmission peak, DT, and spectrum shape. The FWHM depends on statistical fluctuations of the scintillation light captured by the shaper, and therefore, it is not compensated by the gain stabilization. Because the FWHM is inversely proportional to the square root of the number of photoelectrons, the transmission-peak region of interest (ROI) should be selected for the maximum expected temperature where the statistical fluctuations are maximum. If the ROI selected is too narrow, the temperature dependence of the FWHM will affect the net count rate of the transmission peak. Note that this effect is more pronounced at low energies where the resolution is determined mostly by photoelectron statistics rather than by light nonlinearity.

The enrichment measurement is based on the absolute values of the transmission- and uranium-line count rates and, therefore, it relies on the proper DT correction. We have found that the main temperature dependence on counting information is caused by unrecognized pulse pileups [8]. Most modern MCAs have two shapers: slow (spectroscopy) and fast (auxiliary). The gain stabilization feedback loop compensates for the temperature drift of the slow channel (0.5-μs SG line in Fig. 1), but because the fast channel has a different shaping time, the fast-channel pulse-height spectrum will have a different temperature behavior, as indicated by the 0.1-μs SG line in Fig. 1, and it is not stabilized. Therefore, the temperature will cause a change in the fast-channel event-recognition threshold equivalent to the difference between the 0.5-μs and 0.1-μs lines in Fig. 1. This will affect the pileup rejector efficiency in a very complex way as both the pulse-pair resolving time and the number of events seen by the fast channel become temperature dependent. The total effect of this temperature dependence is difficult to compensate by processing of the spectral data because the pileup has a nonlinear count-rate dependence, which interferes with both the transmission and uranium peaks.

The reason why the pileup rejector of a NaI(Tl) spectrometer has a poor efficiency is illustrated in Fig. 2. The rise time of the preamplifier signal shown in yellow is determined by the duration of the detector light pulse. The fast-channel pulse, shown in magenta, is longer than the scintillation light pulse because it results from the convolution of the light pulse and the fast-shaper response to a step function. Therefore, the duration of the fast-channel pulse (Tpulse) is comparable with the pileup inspection interval (Tpeak) in the slow channel (blue trace). The relative percentages of unresolved pileups in the fast channel and slow channel are comparable, and this results in poor pileup rejection efficiency. We stress that the pileup rejector efficiency and its temperature dependence are attributed to the detector signal rather than to the MCA shaper parameters.

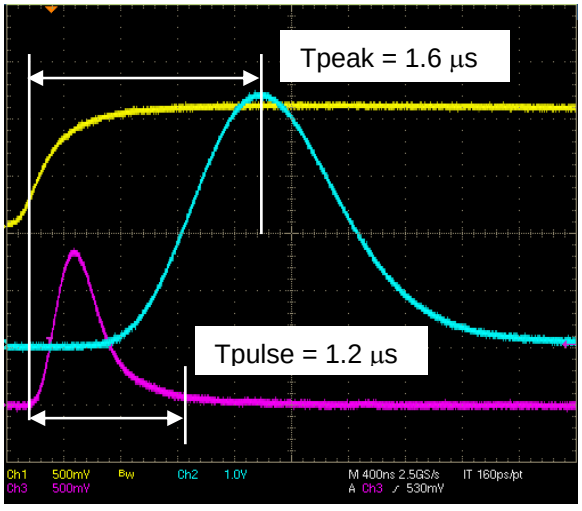


**Fig. 2:** Timing diagrams of pulse shapes for a commercial NaI(Tl) detector with a standard preamplifier and an Ortec 672 spectroscopy amplifier with a 0.5-μs shaping time.

The measured temperature dependence of the net area of the 122-keV $^{57}$Co peak for a detector with a commercial preamplifier is shown in Fig. 3. The temperature was cycled +/–10 $^{o}$C from room temperature at a rate of 2 $^{o}$C/hr. Data were collected and displayed with Trend Analysis software and

the BDMS (blend-down monitoring system) data-acquisition system [9]. The stabilizer gain adjustment was used to indicate the temperature changes. The higher gain adjustment for temperatures above +25 $^{o}$C corresponds to the steeper slope of the peak shift in Fig. 1. The increased duration of the light pulse for lower temperatures causes the bumps in the DT. The 20 $^{o}$C temperature change causes an approximately 0.4% change in the count rate of the transmission peak, which corresponds to a large change in the multiplication factor (Table 1) and the enrichment.

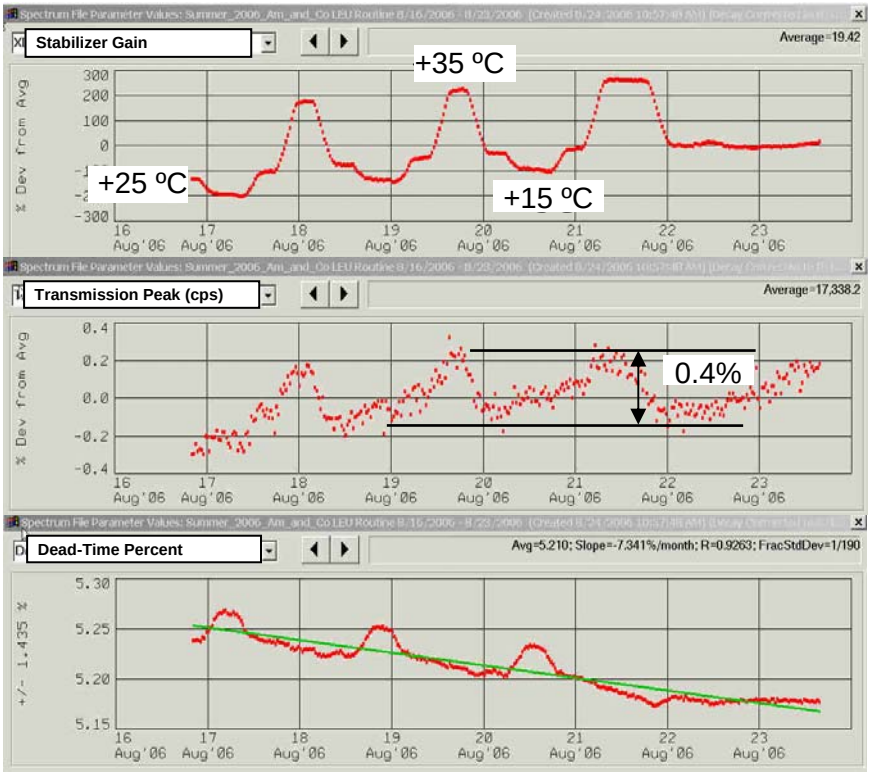


**Fig. 3:** Plots of the stabilizer gain, relative transmission-peak count rate (cps), and DT for a NaI(Tl) detector with a commercial preamplifier.

In our previous work, it was shown that a proprietary shaping technique in the preamplifier improves the count rate and pileup compensation significantly [10]. The implementation of this proprietary technique compensates for the effect of a microseconds-long light pulse on the preamplifier pulse shape (Fig. 4). Small undershoots in both the fast and slow channels have been created to hide the fluctuations of the light-pulse tail below the baseline and improve the pulse-pair resolving time. Based on that design, we have developed temperature compensation for the NaI(Tl) detector that keeps the pulse shape constant over the temperature range, 25 to 45 °C, thus cancelling the temperature dependence of the counting information.

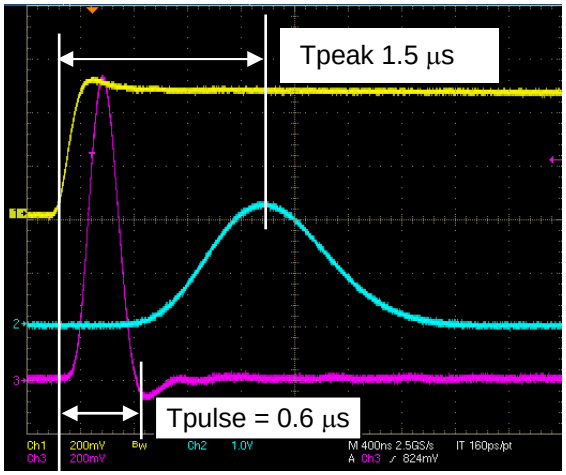


**Fig. 4:** Timing diagrams of pulse shapes for a NaI(Tl) detector with a step-response preamplifier. The fast-channel pulse is half as long as the corresponding pulse in Fig. 2

Because of the complex character of the temperature dependence (amplitude and duration of the fast pulse), an empirical iterative procedure was used to determine the values of the temperature-correction network. The results of the third iteration are shown in Fig. 5. The intensity of the $^{57}$Co total spectrum count rate after background subtraction was about 14,000 cps. The temperature was cycled for 10 days between +25 °C and +45 °C. We did not observe any obvious dependence in the intensity of the transmission peak, but there is still a very small pattern in the fast-channel input count rate, as evidenced by the DT percent. We did similar tests at low count rates and did not find a pattern in the transmission-peak count rate. A series of detectors built using a generic set of temperature-compensating parameters showed a <0.1% change in the transmission-source peak count rate over this temperature range, corrected for the source decay.

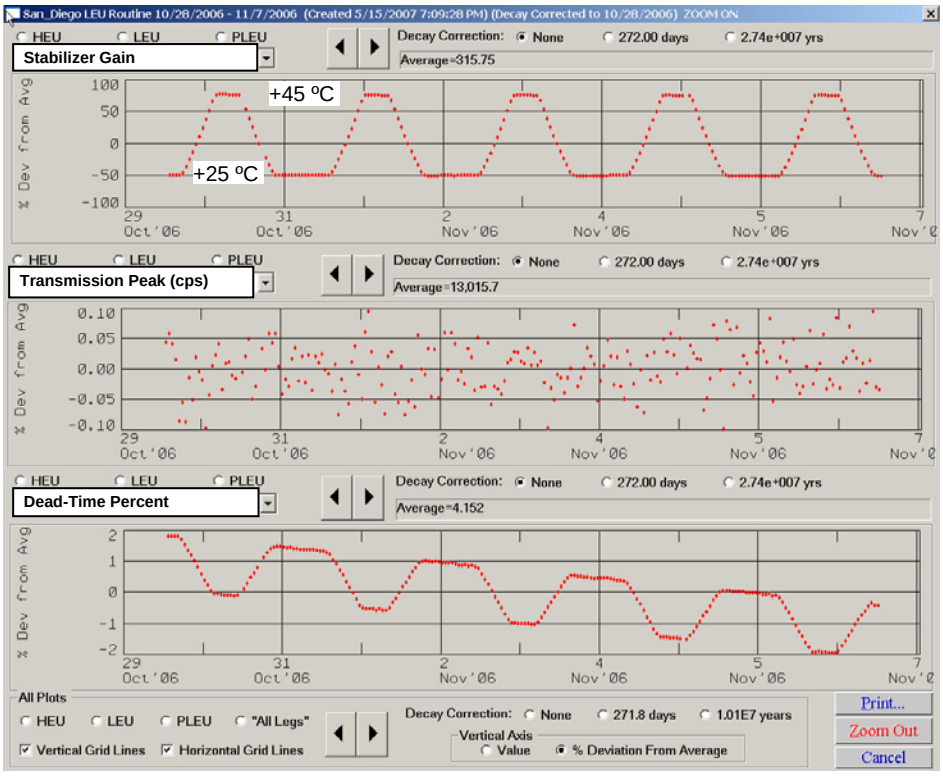


**Fig. 5:** Results from a 10-day test of a temperature-compensating preamplifier with a $^{57}$Co transmission source. From top to bottom: stabilizer gain values for temperature swing from +25 °C to +45 °C, intensity of the transmission peak, and DT percent reported by the MCA.

We are sensitive to photomultiplier-tube gain drift that results from the fatigue of the dynode material because our detectors are meant to operate many years at significant count rates. Therefore, we operate them at a significantly lower voltage than the typical voltage (approximately 600 V versus 1000 V nominal voltage) in order to keep the gain degradation to less than a couple of percent per month.

The temperature stability of the geometry is also important. Aluminum, which is the typical structural material for detectors, has a temperature expansion coefficient of 22 ppm/°C. In our EM, this can cause an error of 0.08% for a 20 °C change in temperature. Note that the geometry effect is not count-rate dependent and, therefore, the unrecognized pileups need to be treated separately. A thermo-compensated geometry of two materials with different temperature coefficients, as in a mechanical pendulum, provides a simple solution to the problem.

## 3. Noise sensitivity of the enrichment results

Even though electrical noise might not affect the intrinsically poor energy resolution of a NaI(Tl) spectrometer, it can affect the DT and, thus, the enrichment results because all counting information scales with the DT. Most of the commercially available detectors and MCAs are intended to work in laboratory conditions; therefore, their noise immunity is usually not good enough to operate properly in a harsh industrial environment. In our EM, the detector and MCA are in a cage suspended from the $UF_6$ pipe, as shown in Fig. 6.

**Fig. 6:** Typical configuration with the monitor connected to the $UF_6$ pipe. The cage with instrumentation is connected to the pipe by the end brackets; the middle bracket carries the source.

The instrumentation inside the cage is connected through 120-V AC power and data communications cables to a remote computer cabinet. For safety reasons, the cage is connected to the system ground at the power outlet. The MCA data output is connected through a 3-ft USB (universal serial bus) cable to the optical communication cable. (The USB-to-optical adapter at the MCA was originally powered through copper wires in the optical cable, causing an additional ground-loop from the MCA to the power outlet.) The $UF_6$ pipe is also connected galvanically or capacitively to the cage, creating a classical ground loop, as shown in Fig. 7. Our observations in the facilities show evidence of switching noise penetrating into the spectrometer and affecting the counting information by the way of the DT correction. The switching noise comes as a burst of bipolar pulses with high frequencies (HFs). One of the particular sources of noise was an HF pickup from stepping motors. Galvanic insulation of the detector and cage reduced the noise but did not eliminate it because it penetrated through the stray capacitance between the detector and cage. The noise affected the DT without any observable changes of the pulse-height spectrum. We have found two major types of noise penetration:

1. Noise induced in the preamplifier through the high voltage (HV) power cable. Some commercial detectors do not have HV filters in the detector base. The effect of that noise is very severe because it is amplified by the preamplifier. Adding a low-pass HV filter prevents that noise.

2. Noise induced in the signal cable between the detector and the MCA input. Even if all cables connecting the detector and MCA are shielded, the noise through the ground loop penetrates and is amplified in the MCA's front end.

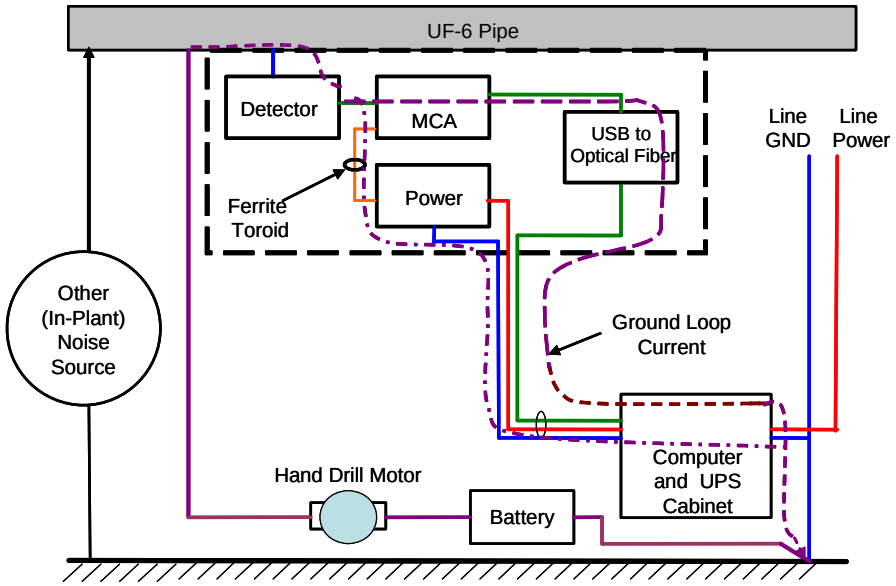


**Fig. 7:** Typical configuration of electrical connections between EM components. The 12-V battery and the brushed motor connected between the cage and system ground introduce noise into the system. Noise may also be introduced by an external source connected between the system ground and $UF_6$ pipe.

The snapshots of the fast and slow channels shown in Fig. 8 can explain the noise effect inside the MCA. The amplitude of HF bipolar noise pulses are suppressed in the slow channel and have amplitudes that are a fraction of the detector signal pulses. Therefore, they are not analyzed and do not affect the pulse-height spectrum. The ratio of the amplitudes in the fast channel is just the opposite; the burst of noise pulses causes multiple triggering for the duration of the peaking time of the slow pulse. This causes triggering of the pileup rejector circuitry, providing incorrect DT to compensate for the nonexisting loss of detector pulses. This phenomenon, known as "hidden DT," is the typical response of the spectroscopy system to HF noise.


**Fig. 8:** Amplitude relationships between signal and noise pulses in the slow and fast channels of the MCA.

To simulate and study the vulnerability of the EM to noise, a mockup system was assembled. A DC battery and brushed motor were connected between the ground line of the AC power outlet and the $UF_6$ pipe. Thus, the motor noise was forced through the ground return of power and data communication cables between the cage and the computer cabinet. By controlling the current through the brushed motor, we were able to reproduce, in a controllable manner, different levels of noise and to study the noise immunity of the spectrometer. We have found that the electrical connection through

the USB cable is more critical for the noise than the one through the DC power cable. One possible reason could be that the MCA DC power input has a built-in common-mode noise filter. For many reasons (not all of them technical), it is not always feasible to insulate the detector and cage from the $UF_6$ pipe; therefore, we have focused on improving the noise-immunity measures between the detector and MCA.

Our strategies for noise reduction were as follows:

a) To enforce the grounding and shielding of the cables between the detector and MCA, thus providing very low impedance for the ground-loop currents. We have installed all input cables, including the DIM, inside a 2-in.-wide copper- braid. The braid was clamped to the detector housing and the input connector of the MCA (Fig. 9).

b) To reduce the ground-loop currents by increasing the impedance for HF noise and by removing unnecessary electrical connections between the MCA and the outside world. The USB cable was replaced with a very short one, and the USB-to-optical adapter was powered from the DC power supply inside the cage. This removed the galvanic connection between the MCA USB connector and the power outlet. A toroid was installed on the DC power cables for the MCA and the optical-cable adapter for additional HF decoupling.

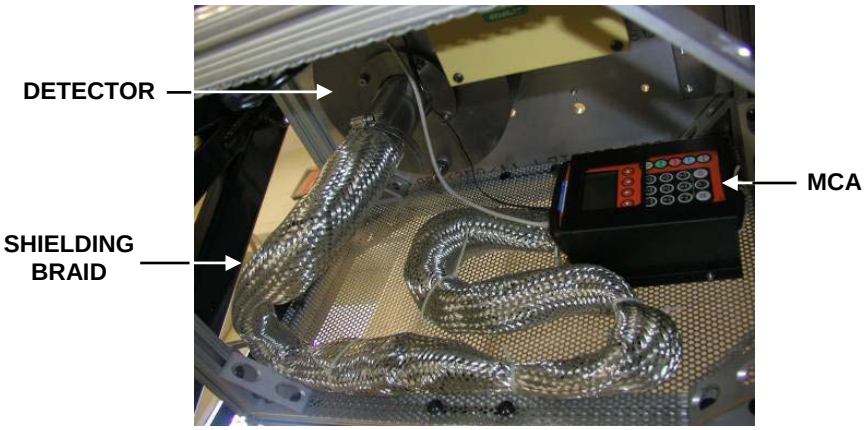


**Fig. 9:** Shielding configuration in the cage: enforced ground and shielding between the detector and MCA; ferrite decoupling of power and USB signal cables.

For an extreme test, the EM cage was connected to a welding table by a 0.5-in.-wide copper strap (Fig. 10). The data-collecting computer and interruptible power supply were 30 ft apart and were connected to the cage through power and data communication cables. Low-count-rate spectra were collected while the welder was operating and again while it was off, and the DTs were found to be equal to within 0.01%. However, the ultimate test was done with the HV turned off and the lower-level discrimination set at the beginning of the noise slope of the fast channel (in this case, channel 3). The reading of the fast channel was used as an indicator of whether the welding noise penetrated the MCA. Again, no difference was observed between data collected when the welder was turned on or off.

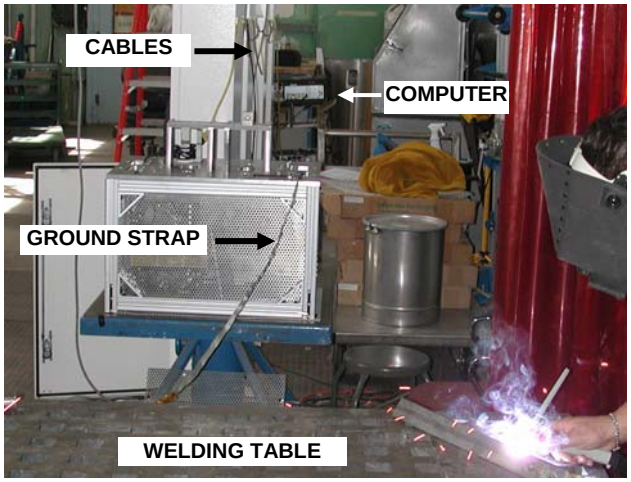


**Fig. 10:** Configuration of the welding experiment. The cage is connected to the welding table using the copper strap. The ground loop is created by the power cable and computer back in the room. The welding noise did not affect either the NaI spectrometer or the communications between the MCA and computer.

## 4. Results of long-term testing

A mockup EM system was assembled at Los Alamos National Laboratory and run with discrete sources ($^{57}$Co and $^{235}$U) over a long period of time to test the difference in stability between a standard setup and the new improved setup. The standard setup was installed in a cage, as shown in Fig. 6. It included an off-the-shelf NaI(Tl) detector, an MCA, a 12-V power supply for the MCA, and a fiber-optic cable for communicating back to the computer, which was located outside of the cage. The fiber-optic cable was 30 meters long and contained copper wires within to provide 5-V power from the computer end to the transmitter/receiver module connected to the MCA inside the cage. The improved setup was also installed in a cage, as shown in Fig. 6. This setup is the same as the standard setup except the detector has been upgraded with our new patented technology preamplifier, discussed in Section 2, and the detector and MCA have been grounded together using the 2-in.-diameter copper-braid shield shown in Fig. 9. (Ferrite toroid decoupling was not used.)

Figure 11 shows the enrichment percent, transmission net count rate ($^{57}$Co), and the DT percent from both of the setups over approximately a 1-1/2 month period. A few conditions were changed during this time period to test the systems' response to different levels of noise interference from a stepping motor. To start, both systems ran without any noise added from the stepping motor. On the date indicated in Fig. 11 by arrow 1, the stepping motors were plugged into the same AC power circuit as the two systems to test if the noise could penetrate into the systems through the circuit grounding. Arrow 2 in Fig. 11 indicates when the stepping-motor noise was more strongly coupled into each of the cages using a 1-in. braided-copper strap between the motor apparatus and each cage. Initially, the standard setup would not operate with such strong coupling to the noise, so the fiber-optic cable of that system was upgraded to eliminate the copper wires in it and to power the module in the cage directly from an Acopian power supply. The system then worked, and the data displayed in the figure, starting at arrow 2, comes from this configuration. The fiber-optic cable on the improved system was also upgraded.

As displayed in Fig. 11, the improved system outperforms the standard system with respect to noise immunity. The enrichment percent and DT percent values from the improved system are not influenced by the stepping motors, whereas those in the standard system are influenced. The standard system produces inaccurate and less precise enrichment values even though the enrichment has been constant throughout the measurement period.

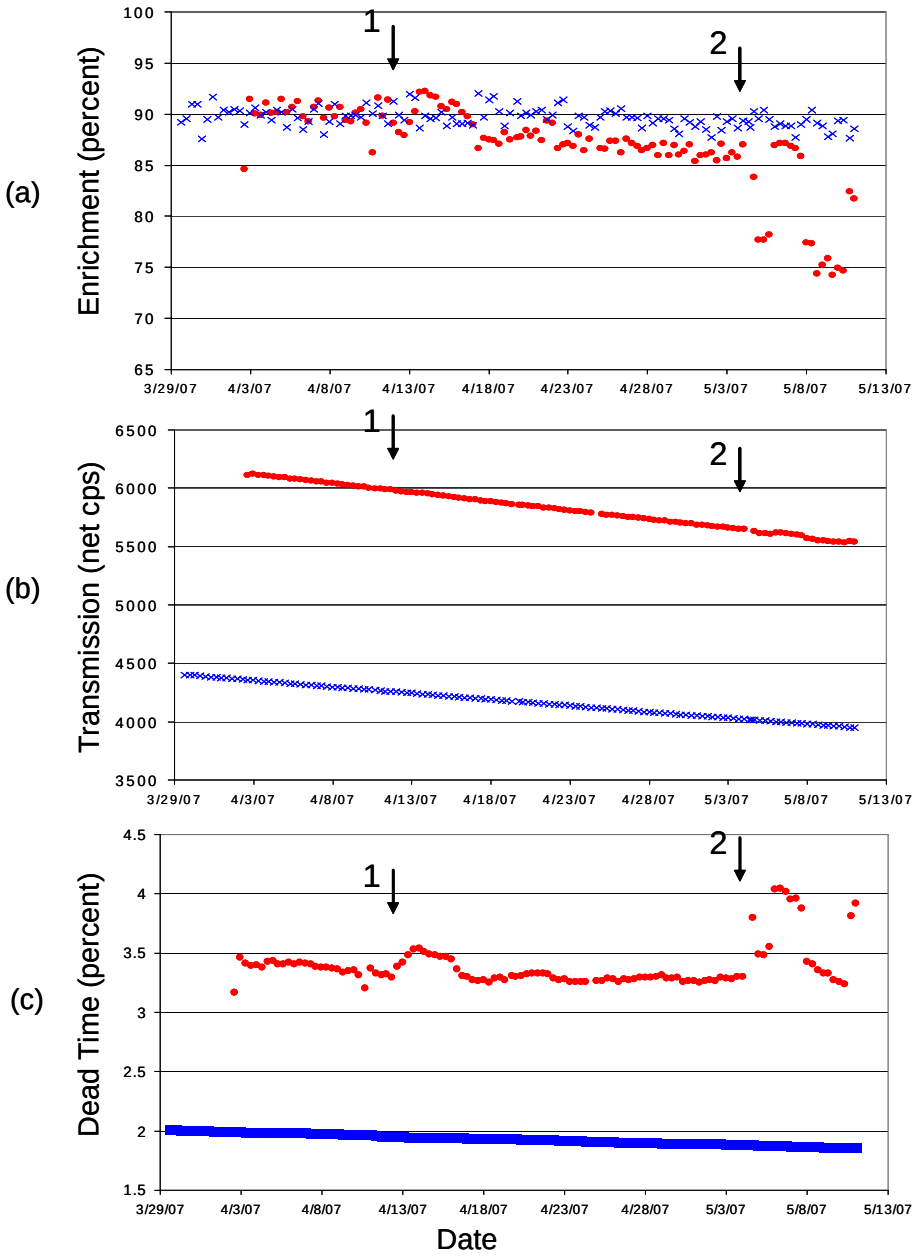


**Fig. 11:** Results from the long-term tests on standard EM setup (red circles) and the improved EM setup (blue Xs). Graph (a) shows the calculated enrichment percent data, (b) shows the transmission net cps, and (c) shows the DT percent.

## 5. Systematic errors due to spectral-line interference

The low resolution of scintillation detectors presents another challenge—the processing of the spectral information. The spectral lines of interest overlap. The background under each peak represents a significant fraction of the net area of the peak and, therefore, its subtraction introduces significant systematic and random errors. The shape of the spectrum is changed by changing the ratio between the intensities of the transmission and $^{235}$U sources as well as by changing unrecognized pileups due to the decay of the transmission source. To estimate the effect, we mounted a $^{57}$Co source in a fixed position in front of a detector and illuminated the detector from the back with different intensities of $^{235}$U sources so that changing the $^{235}$U source did not affect the $^{57}$Co geometry. The intensity of the 122-keV peak net area was set to 3,600 cps, and the intensity of 186-keV peak net area was initially set to 360 cps. The intensity of the 186-keV peak was varied down to 60 cps, which gives a 122-keV to 186-keV ratio of 60 and corresponds to a fresh $^{57}$Co source with approximately 20,000 cps in the 122-keV peak net area (Fig. 12).

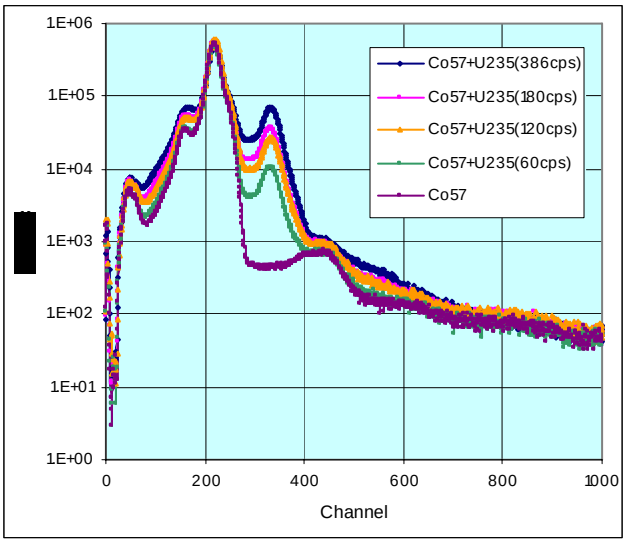


**Fig. 12:** Overlapping $^{57}$Co and $^{235}$U spectra. Even if the $^{235}$U intensity is only a small fraction of the $^{57}$Co intensity, the variable intensity of the $^{235}$U spectrum significantly changes the background of the 122-keV $^{57}$Co peak.

The relative changes of the calculated net areas from the data in Fig. 12 are tabulated in Table 2. One can see that a factor of 2 change in the intensities ratio, which corresponds to the 272-day decay half-life of $^{57}$Co, leads to a 2.1% error in the calculated transmission-peak net area. Note that the effect of the pileup and proper DT correction are not taken into account

| 186-keV Peak Intensity (cps) | 122-keV Peak Relative Intensity (%) |
|---|---|
| 386 | 97.44 |
| 180 | 98.89 |
| 120 | 99.55 |
| 60 | 99.95 |

**Table 2**: Relative change of the net area of the 122-keV peak.

## 6. Conclusions

The effects of environmental factors, such as temperature, pickup noise, and geometry stability, on the measured enrichment have been evaluated. By using a patented technology in the NaI(Tl) detector preamplifier, we are able to reduce the temperature error below the statistical fluctuations of the transmission-peak count rate and improve the count-rate characteristics of the NaI(Tl) spectrometer. With proper shielding and grounding, we were able to improve the noise immunity enough to tolerate an operating welder connected to the cage. Some of the systematic errors related to the calculation of the transmission-peak intensity have been evaluated.

The experience gained toward improving the stability and accuracy of the classical EM will be applied to the development of the next generation of EM, which will have enough sensitivity to measure low-pressure gas without penetrating the pipe and will not need frequent source changes.

## 7. Acknowledgements

This research was supported by the National Nuclear Security Administration, Office of Defense, Nuclear Nonproliferation. We acknowledge the generous technical and financial support we received from Kurt Siemon, Janie Benton, Elena Thomas, Dennis Meyers, Dave Dougherty, and Dan Decman in the Dismantlement and Transparency Office and from Tom Hill at the Los Alamos National Laboratory. Without their participation, this work would not have been possible.

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# A Pulsed Neutron Device for Studies of Non-nuclear Material Detection and Fissile Material Assay

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***Abstract:***

*The Joint Research Centre has taken into operation a new experimental device designed for research in the fields of nuclear safeguards and illicit trafficking. The research projects currently undertaken include detection of shielded contraband materials, and mass determination of small fissile materials in shielded containers. The device, called the Pulsed Neutron Interrogation Test Assembly, incorporates a pulsed 14-MeV neutron generator and a large graphite mantle surrounding a sample cavity. In this configuration a relatively high thermal neutron flux with a long lifetime is achieved inside the sample cavity. By pulsing the neutron generator, a sample may be interrogated first by fast neutrons only, and a few hundred micro-seconds later by a pure thermal neutron flux. For the detection of materials such as conventional explosives the device employs scintillation detectors for the detection of characteristic prompt gamma rays either from inelastic scattering by fast neutrons, or from capture of thermal neutrons. The paper reports on the design of the new device, the characteristics of the pulsed fast and thermal neutron source, and the current activities for the project of non-nuclear material detection.*

**Keywords:** neutron generator; illicit trafficking; activation; NDA; nuclear safeguards, nuclear measurement techniques; pulsed neutrons

## 1. Introduction

The Joint Research Centre of the European Commission develops instrumentation and analysis methods for the non-destructive assay of nuclear materials and for detection of contraband materials. In relation to this, a new experimental device was designed and constructed in the laboratory of the *Institute for the Protection and the Security of the Citizen* (IPSC). The device, called the Pulsed Neutron Interrogation Test Assembly (PUNITA), incorporates a pulsed (D-T) neutron generator. As such, the instrument finds applications in many different areas although the main research activities concern NDA methods for nuclear safeguards, and for detection of illicit trafficking of nuclear and non-nuclear materials. The nuclear safeguards application of PUNITA concerns the determination of the mass of fissile material in a sample independent of matrix materials and spatial source distribution. To achieve this aim, intense bursts of thermal neutrons, produced by the neutron generator and a substantial graphite liner, induce fission in the fissile isotopes of the sample. The fission neutrons are detected and analysed in the time domain. The analysis method consists of a further development of the standard passive neutron correlation technique [1]. Another research project currently undertaken in PUNITA concerns the detection of contraband materials such as drugs and explosives. For this purpose the methods being investigated apply the detection of characteristic gamma radiation from inelastic scattering reactions by fast neutrons and neutron capture by thermal neutrons on specific elements. An interesting possibility currently being studied in PUNITA consists of combining both inelastic scattering and capture in the same measurement. For this purpose neutrons emitted from the pulsed neutron generator interrogate the sample, first by fast neutrons directly from the neutron generator, and a few hundred microseconds later by thermal neutrons produced in the surrounding graphite liner. The resulting characteristic gamma lines are analysed to yield element ratios of the sample under interrogation [2].

## 2. Description of the Instrument

The Pulsed Neutron Interrogation Test Assembly (PUNITA) is a versatile experimental tool. The instrument is designed to give the experimenter maximum flexibility with respect to detector and sample arrangements. When in the closed configuration the instrument forms a cube with a central void called the sample cavity (Figure 1). The top and bottom sides of the cube are part of a central structure, while the four vertical sides are located on movable trolleys. The trolleys move on a rail system to yield a perfect mating with the central structure. The size of the sample cavity is 50 cm by 50 cm cross-section and 80 cm height. The accelerator assembly of the (D-T) generator has a length of only 43 cm and can be placed without constraints anywhere inside the cavity. A thick graphite liner is located on all six sides of the sample cavity. On the vertical sides, the graphite liner is integrated in the trolley to give access to the cavity from all sides when in the open configuration. A total of 1,350 kg of reactor grade graphite is used in the liner.

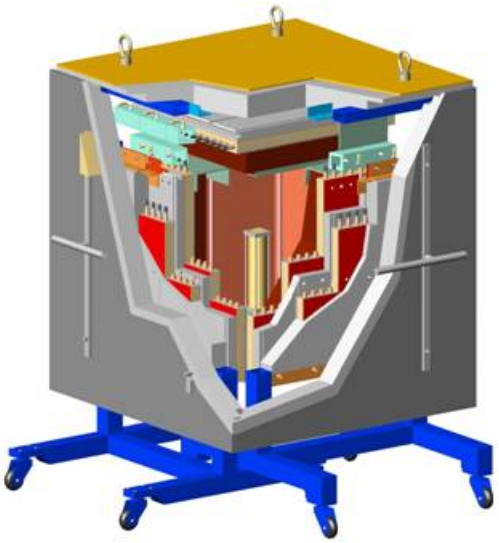

**Figure 1: The JRC Pulsed Neutron Interrogation Test Assembly (PUNITA)**

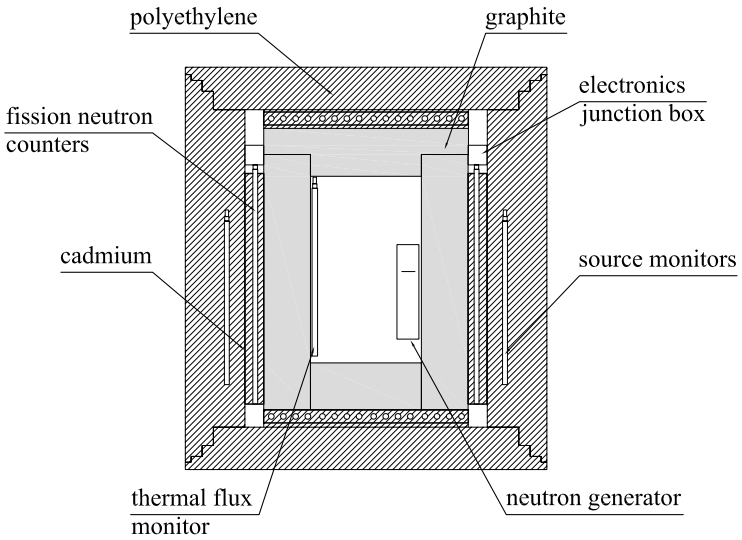


**Figure 2: Cross section of the PUNITA facility. The different components of the assembly are indicated.**

A fast neutron detector module is integrated in each of the six sides immediately behind the graphite liner. The modules, nominated fission neutron counters, include sixteen $^{3}$He proportional detectors of 3040 torr and 1000 mm length each. The $^{3}$He detectors are embedded in a single row in a block of polyethylene with a 1 mm cadmium cladding on all surfaces. A neutron shield of 300 - 350 mm polyethylene is placed behind the fission neutron counters on all six sides. On each of the vertical sides eight $^{3}$He neutron detector of 3040 torr and 500 mm length, nominated source monitors, are embedded in the polyethylene shield. Other permanently installed instrumentation include bare $^{3}$He counters of pressure below 760 torr, nominated thermal flux monitors, located in the corners of the sample cavity (see Figure 2).
The neutron generator, a model A-211 from Thermo Fisher Scientific Corp., is capable of emitting 14-MeV neutrons at a rate of $2 \cdot 10^{8}$ s$^{-1}$. The generator can be pulsed at rates from 10 Hz to 150 Hz. The duration of the 14-MeV neutron pulse is about 5 μs resulting in a duty cycle as small as 0.5 per mille at 100 Hz pulsing. This feature permits a very good time separation of the fast and thermal neutron interrogation in PUNITA. Another important feature of the generator is the pulsing of both the Penning ion source and the acceleration voltage assuring that no neutrons are emitted between pulses. The 14-MeV neutrons emitted in each generator pulse are thermalised in the graphite liner in a period of about 200 μs following the pulse. After this time only thermal source neutrons persist in the cavity. After reaching a maximum value at about 280 μs after the fast neutron pulse, the thermal neutron flux in the sample cavity decays according to a single exponential function [3]. In this time period a fissile sample in the cavity would undergo fission by thermal neutrons only. Also in this period, only the fast fission neutrons are detected in the cadmium covered fission neutron detectors. These features are essential in the projects concerning fissile material assay. The source monitors located in the shielding are used for normalisation of the 14-MeV neutron emission from the generator. Likewise, the bare $^{3}$He detectors in the sample cavity are used for normalisation of the interrogating thermal flux. The response of the various neutron detectors mentioned here are all recorded with Ortec MCS multi-channel scalers which are triggered synchronous with the pulsing of the neutron generator.

The research projects currently undertaken in PUNITA require a good determination of the interrogating thermal and fast neutron flux. To achieve this, we have measured the response of both a small $^{3}$He proportional counter, a $^{235}$U fission chamber, and activation foils [3]. The response of the small $^{3}$He detector and the fission chamber had been normalised in terms of count rate per unit thermal flux using a standard set-up composed of a large solid block of high density polyethylene and a calibrated (energy spectrum and neutron output) AmLi neutron source. Both MCNP [4] calculations and measurements with and without cadmium covers confirmed that the epi-cadmium response in the standard set-up was much less than 1% of the thermal response. When placed in the centre of PUNITA, the detectors were measured both with and without cadmium in order to subtract the substantial fast neutron response in the first 200 μs after the 14-MeV neutron pulse. The resulting time response of the thermal flux in the centre of the sample cavity, as recorded with a multi-channel scaler, is shown in Figure 3. The fission chamber and the small $^{3}$He detector response functions were identical. Figure 3(a) shows a maximum thermal flux at 280 μs after the fast pulse.

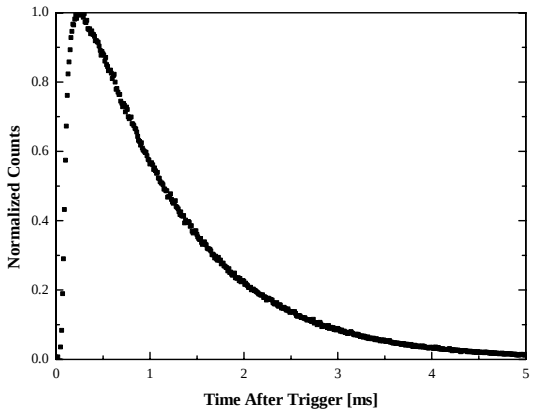


Fig.3(a)

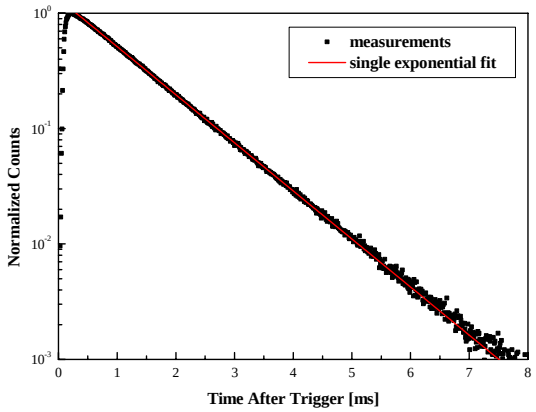


Fig.3(b)

**Figure 3: Thermal neutron pulse as produced by the neutron generator and the graphite moderator. Figure 3(a) and 3(b) present the thermal flux behaviour measured by means of the fission chamber and the He-3 tube respectively. Figure 3(b) shows single exponential fit to the decay curve.**

Figure 3(b) shows the thermal flux response in logarithmic scale. The decay of the thermal flux is clearly a single exponential function. The solid line represents a least-square fit to the decaying curve. The reciprocal of the decay constant (i.e. the thermal neutron lifetime) is calculated to 1.05±0.02 ms.

The average thermal flux, derived from the counting rate in the two detectors, gave the values $6.51\pm0.39\cdot10^3$ $s^{-1}cm^{-2}$ and $6.15\pm0.37\cdot10^3$ $s^{-1}cm^{-2}$ from the small $^3$He detector and the fission chamber, respectively, at a 14-MeV neutron emission rate of approximately $0.7\cdot10^8$ $s^{-1}$. In both cases the largest component to the error is estimated to be the positioning of the detector in the standard set-up. The measurements with the two gas counters were confirmed by foil activation measurements. We used niobium foils to determine the 14-MeV flux and gold for the thermal flux. The epi-thermal neutron activation was eliminated by cadmium ratio measurements. The foil measurements agreed with the gas counters within the error margins [3].

## 3. Research activities

As PUNITA is a relatively new research tool much of the research activities so far have concentrated on the characterization of the pulsed neutron source and the detection systems. The fact that both the fast and thermal neutron source term is well characterised makes PUNITA an interesting research tools for many different applications. Some specific requests concern neutron and gamma dosimetry, and failure of electronics under fast and thermal neutron exposure. In the present paper we only discuss the status of the two most important research activities: fissile material assay, and detection of contraband materials.

### 3.1. Fissile material assay

The nuclear safeguards application of PUNITA concerns the determination of the mass of fissile material in a sample independent of matrix materials and spatial source distribution. To achieve this aim, intense bursts of thermal neutrons, produced by the neutron generator and the graphite liner, induce fission in the fissile isotopes of the sample. The analysis method consists of a further development of the standard passive neutron correlation technique. One of the analysis methods we are currently investigating is outlined in reference [1,5]. The method is particularly effective for small amounts of fissile material and is expected to be capable of assaying $^{235}$U and plutonium at unprecedented low quantities (< 1 mg) making the method particularly suitable for example for waste characterisation. Another safeguards related application being investigated in PUNITA is the fissile material diversion scenario where plutonium is substituted by curium. In this method the ratio of thermal induced fission (in either $^{245}$Cm or $^{239}$Pu) to spontaneous fission (in either $^{244}$Cm or $^{240}$Pu) is used as a fingerprint. An important component for the fissile material assay is the new frequency analyser (multiplicity counter) capable of dealing with the particular triggering requirements of observation intervals and synchronisation with the pulsed neutron source which cannot be achieved with a standard multiplicity counter. For this purpose we have developed a fast timestamp and list mode based analyser which is currently undergoing tests in our laboratory [5,6].

### 3.2. Detection of contraband materials

A wide range of methods have been proposed in the past for automated, non-destructive detection of hidden dangerous/contraband materials located inside consignments of luggage or goods. In the PUNITA facility we study the methods which apply the detection of characteristic gamma rays subsequent to neutron irradiation. Many types of neutron reactions will cause the emission of characteristic gamma rays from a target material. Among these reactions, known to the method of prompt gamma activation analysis (PGAA), are in-elastic scattering (n, n′$\gamma$) by fast neutrons and the neutron capture reaction (n, $\gamma$) by thermal neutrons. The gamma energy from such reactions is characteristic for the chemical elements present in the sample. By gamma spectroscopy it is possible to identify elements in the target material and to some extend the element composition by determination of gamma line ratios. An interesting possibility currently being studied in PUNITA consists of combining both inelastic scattering and capture in the same measurement. For this purpose neutrons emitted from the pulsed neutron generator interrogate the sample, first by fast neutrons directly from the neutron generator, and a few hundred microseconds later by thermal neutrons produced by the surrounding graphite liner. Another objective in our research is to identify

and test instrumentation which can be applied easily in an industrial environment without excessive maintenance and experts' interventions. Although, we have used a HPGe detector in the first experiments to identify the numerous peaks in the gamma spectra, a scintillation detector with low maintenance would be a desirable. For this reason we test a newly developed scintillation detector based on the $LaBr_3$:Ce crystal in parallel with the HPGe. This detector offers a substantial advantage over the standard NaI(Tl) detector such as energy resolution (about 3 % at 662 keV), and high efficiency for gamma rays in the MeV range. The detector crystal has other advantages such as a high scintillation light output with a fast decay time. Our first investigations concerned the behaviour of the lanthanum bromide detector alone and in conjunction with the neutron generator. The biggest drawback of the $LaBr_3$ detector is the internal radioactivity. The internal radioactivity is due to naturally occurring radioisotopes $^{138}$La and $^{227}$Ac [7]. The radionuclide $^{138}$La has two gamma-lines at 788.7 and 1435.8 keV. Actinium is chemically very similar to lanthanum. Both elements have been reduced during the production of the scintillation crystal, but they can still affect the background. Figure 2 shows a background spectrum from a 38.1mm x 38.1 mm $LaBr_3$:Ce scintillation detector. The spectrum was acquired inside the PUNITA facility. For this reason the naturally occurring radiation from $^{40}$K is eliminated

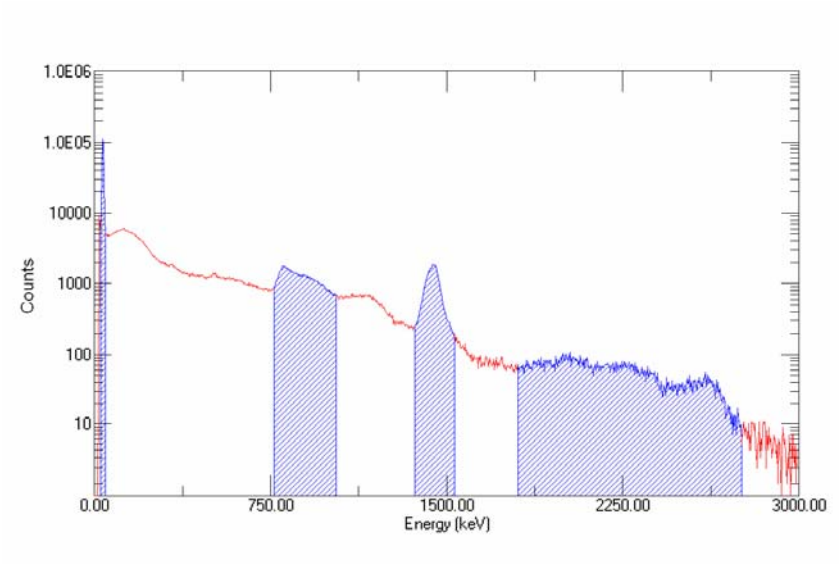


**Figure 4: background spectrum recorded with the LaBr3:Ce scintillation detector**

The internal contamination of the lanthanum bromide detector that we observed is confirmed in other literature such as reference [7]. The values are given in Table 1.

| Energy range (keV) | Contaminant | Count rate-this work (cps) | Count rate –Ref. [7] (cps) |
|---|---|---|---|
| 30 – 40 | X-rays from $^{138}$La | 5.0 | 18 |
| 750 – 1500 | 789-keV β and γ from $^{138}$La | 2.3 | 2.6 |
| 1430 – 1470 | 1436-keV γ from $^{138}$La | 2.3 | 3.1 |
| 1500 – 3000 | α's from $^{227}$Ac | 0.4 | 1.2 |

**Table 1: Count rates from internal contaminants in $LaBr_3$:Ce scintillation detectors**

For the purpose of identifying drugs and explosives we intend to apply the prompt gamma activation analysis for the determination of element ratios. The characteristic gamma radiation is typically in the MeV range as for example hydrogen (2.23 MeV), carbon (4.44 MeV), nitrogen (10.8 MeV) and oxygen (6.13 MeV). For this reason we needed to obtain a full-energy efficiency calibration of the lanthanum bromide detector in this range. For this purpose we use standard gamma sources such as natural thorium but also activated sources. We produced the short-lived $^{66}$Ga isotope ($T_{½}$ 9.4 hours, 4.806 MeV γ-line) by the (p,n) reaction on $^{66}$Zn using the Cyclotron of the Joint Research Centre in Ispra. The calibration curve is presented in reference [8].

Another requirement for obtaining useful activation gamma spectra is a good shielding of the detector. Due to the close vicinity of the neutron generator to the lanthanum bromide gamma detector in PUNITA, we need to shield the detector from the effects of fast and thermal neutrons. We are currently studying a variety of shielding arrangements around the gamma detector. Figure 5 shows a shielding arrangement composed of cylindrical shapes of polyethylene, cadmium liner, tungsten and cadmium liner to surround the lanthanum bromide detector. The investigations to optimise the detector shielding are ongoing.

**Figure 5: Shielding components: (from left) cadmium cup, two tungsten cylinders, further cadmium cup and polyethylene cylinder. The $LaBr_3$:Ce scintillation detector is also shown.**

## 4. Conclusions

The new experimental facility PUNITA has been taken into operation. The initial research activities concentrated on the characterization of fast and thermal flux generated by the pulsed neutron generator and the graphite liner. Two main research activities are currently under development in this facility. One project concern the assay of fissile material by active neutron correlation. The other project concerns the detection of non-nuclear contraband materials. Both project are at an early stage but expected to deliver results in the coming months. The final aim for these projects is to develop the analysis methods, design criteria and performance values for future in-field installations.

## 5. Acknowledgements

The authors are indebted to Dr. R.Berndt for enthusiastic and enlightening discussions and to Mr. J-M Crochemore for the help with setting-up the experiments and his assistance during the measurement campaigns.

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# Session 10

# IS Technical development

# The Application of Geographical Information Systems in Additional Protocol Safeguards

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**Dirk Schmidt, Dialogis GmbH, Germany**
**Bernd Richter, Forschungszentrum Jülich GmbH, Germany**

## *Abstract:*

Countries with an Additional Protocol (INFCIRC/540 corr., AP) in force grant the International Atomic Energy Agency (IAEA) the right to visit and inspect locations where the Agency has a well-founded suspicion of undeclared nuclear activities or materials or where clarification of unresolved questions is needed. Such an inspection scenario is defined as Complementary Access (CA). It is fundamentally different from traditional safeguards where the presence of declared nuclear materials is verified in order to detect a diversion. While traditional safeguards are based on nuclear material accounting, CA inspections follow a more qualitative approach. The inspectors have to look for indicators and signatures in order to detect covert nuclear activities and materials, if they exist. This makes it very important to focus inspections on gathering the appropriate data and information for instant and follow-up analysis. Also, the inspectors have to proceed as efficiently and effectively as possible in acquiring information. For later analysis, it is crucial to log the exact position where samples were taken or measurements were recorded. Global positioning may be a viable solution for outdoor scenarios, but other data need to be tagged to a location within a nuclear facility. One way to conveniently display such data is using Geographical Information Systems (GIS). The following paper will outline the basic principles of GIS and discuss their applicability in safeguards. Then, a number of possible scenarios showing the advantages of data logging and display on a map will be outlined. Next, the integration of GIS data with other AP tools such as satellite imagery, environmental sampling, and wide area monitoring will be presented. Options on how to implement a GIS component into future safeguards instrumentation will conclude the paper.

**Keywords:** inspection, GIS, global positioning, safeguards, nuclear

## Introduction

The implementation of the Additional Protocol (INFCIRC/540 corr., AP) has introduced a significant change in the paradigm of international safeguards and treaty compliance verification efforts. Under the traditional safeguards regime, inspection activities are concentrated on verifying the correctness of declarations of a Non-Proliferation Treaty (NPT) signatory state with a comprehensive safeguards agreement (defined under INFCIRC/153 corr.) in place. Efforts are driven by quantitative concerns and include controlling the correctness of nuclear accountancy, verifying the composition of declared nuclear materials, assuring the integrity of materials under seal, and monitoring activities during International Atomic Energy Agency (IAEA) inspector absence.

The AP, on the other hand, concentrates on the completeness of a state's declarations, thus adding a qualitative dimension to safeguards. With the inspection tool of Complementary Access (CA), inspectors gain the right to visit and inspect locations that are not declared as part of the state's civil nuclear fuel program and where the IAEA assumes to have a well-founded suspicion of undeclared materials or activities or where clarification of unresolved issues is required. Inspectors have to look for indicators and signatures that might hint at undeclared activities or that might give assurance of their absence.

The quality of the information gathered during CAs is of high importance, especially as such inspections are not conducted as stand-alone activities; rather, they are the consequence of the

analysis of various open information sources such as satellite imagery, scientific literature, and Internet. Data management and follow-up analysis to compare existing information with the information gathered during the CA becomes an activity crucial to the success of detecting undeclared nuclear programs.

During CAs, inspectors will use a variety of instrumentation with measurement as well as sample taking and analysis capabilities to further deepen knowledge on leads identified by other information sources and to provide the tools to follow up on suspicions that warrant closer investigation. The exact location where samples and measurements were taken as well as any other location-specific information the inspector deems worthy of notice can be of intense importance during the inspection and when results are analyzed afterwards.

Global positioning system (GPS) data might be of use for outdoor scenarios, but inside facilities data will have to be tagged to their location of origin using different methods. One convenient way to display a broad range of information, including GPS, is the Geographical Information System (GIS) based method. The following paper will outline the basic principles of GIS and reflect on their applicability for safeguards. Then, a number of GIS scenarios, one of which has already been initiated, that have similar advantages as those projected for safeguards, including data logging and map display, will be discussed. The integration of GIS solutions with other AP measures such as satellite imagery, environmental sampling, and wide area monitoring will be presented. Options on how to implement a GIS component into future safeguards instrumentation will conclude the paper.

## Principles of GIS

GIS are computer systems based on hardware and software which allow its users to acquire, store, visualize, analyze, and in that way 'manage' spatial data. There are in general two different kinds of spatial data: raster data (e.g., aerial or satellite images or digital elevation models) and vector data (e.g., location points, street networks, or administrative boundaries), both of which can be processed within GIS. Basic GIS operations allow the calculation of distances and spatial buffers, the classification of remote sensing data, or the aggregation, overlay, or clipping of data based on their geographical position. The system architecture of GIS can range from simple desktop software, mobile GIS clients for PDA or mobile phones, web-enabled tools to use GIS functionality inside the browser as web-services, up to complex geodata infrastructures (GDI) with very specific functionality implemented on the parts of clients and servers.

The wide range of existing GIS-architectures currently offered by software providers has made available a diverse number of processes, data models, and formats. Therefore the consideration of international technical (e.g., XML, Java, IP) as well as conceptual standards provided by the Open Geospatial Consortium (OGC)[1] is the precondition to achieve interoperability between the different GIS software packages. Moreover, the consideration of OGC standards is the key to integrate GIS-functionality in nearly any kind of computer supported system in a standardized way.

This opens the door to use the power of spatial information for the support of different kinds of information management processes. Different levels of GIS-based information management support can be distinguished; the one most frequently used is the visualization of spatial data as 2D maps or 3D scenes. The most well-known application of this kind is Google Earth.[2] The map-based visualization of any kind of geo-related information is a fundamental method for most of the other more sophisticated GIS services like spatial search, orientation, tracking, and navigation.

The analysis capabilities of GIS are already being used extensively at the IAEA, especially processing aerial and satellite images with analysis methods like change detection algorithms to support monitoring tasks and the detection of undeclared nuclear activities. However, other areas of safeguards can profit from GIS applications, too. In the area of traditional safeguards, a first effort of using a GIS tool for the Next Generation Surveillance System (NGSS) has been initiated.

## GIS and the Next Generation Surveillance System

NGSS is intended to be the replacement for the current, standard IAEA surveillance system, the DCM-14. Drawing on the experience made during the lifetime of the DCM-14, NGSS will provide a more versatile surveillance infrastructure that is open to different sensors, standardized data management and review, and addresses the complete lifecycle of the instrumentation. The openness of NGSS towards other sensors is a key feature for the integration of different safeguards disciplines (e.g., non-destructive assay, sealing, and surveillance) and to allow for cross triggering, data correlation, as well as diversity and redundancy.

In this context, the developers of NGSS felt it necessary to provide a tool that would provide the capabilities of displaying the data of multiple sensors and information about them in a more consolidated fashion. Even though inside nuclear facilities the attachment of GPS data is not available, a GIS-based solution was identified as a convenient way to pool data, tag data to specific locations inside facilities, and provide a comprehensive interface that allows inspectors to add, review, and analyze data. The NGSS GIS application is based on a map server that displays, depending on the access rights of the user, all countries an inspector has been accredited to. Selecting a country will drill the map down and display all facilities the inspector is allowed to inspect. Selecting a facility then displays an outline, aerial picture, satellite image, or any other map level the inspector chooses. Inside the facility, the installed sensors are displayed. By selecting a sensor, the inspector can see sensor-specific data such as installation date, maintenance schedule, settings, and a sample image in case of a camera. Figure 1 outlines the basic functioning of the NGSS GIS application.

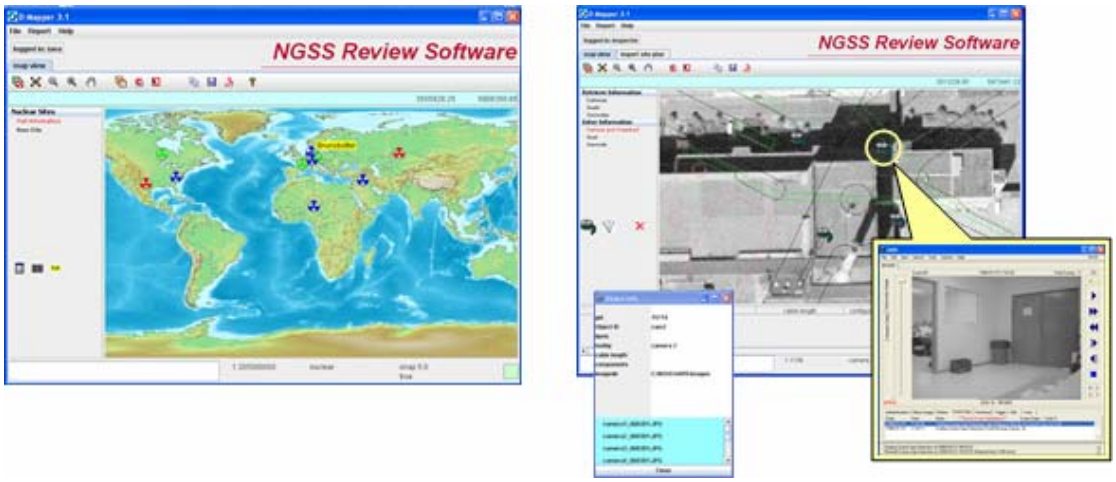


**Figure 1: NGSS GIS Application**

The NGSS GIS application provides inspectors with a spatial overview of the site based on aerial image maps, topographic maps, or floor plans that can be available to him/her during the preparation of the inspection or even on a hand-held computer device while he/she is in the field. The mapping tool supports the inspectors primarily by performing three intuitive, time saving functions.

It provides spatial awareness to inspectors who are not familiar with the site. They can view the layout of the facility and the positioning of the sensors, thereby allowing them to plan their inspection accordingly. Secondly, the map works as an intuitive user interface which gives access to information (e.g., reports, video data, and measurements) related to objects presented on the map. Third, the NGSS map-tool allows the safeguards review of sensor data to be started by clicking on the sensor symbol presented on the interactive map display.

## GIS and Additional Protocol Safeguards

For inspections under the AP, the requirements for instrumentation are fundamentally different than for traditional safeguards. Rather than verifying declared materials and activities, inspectors face situations where they need to investigate materials or facilities for which they have limited knowledge. Supporting instrumentation thus needs to fulfill two major tasks: it needs to be portable, or more specifically hand-held, and it must provide a broad variety of detection functions. Essentially, the goal of AP and CA inspections is to gather sufficient information about indicators and signatures that, in combination with other information sources, can assist in drawing conclusions about the presence or absence of undeclared nuclear materials or activities.

New and advanced hand-held devices, especially in the areas of communication (cell phone, email service, etc.) and data management and display (GPS, GIS, etc.), as well as computing capabilities make these devices natural candidates as an infrastructure backbone for AP instrumentation. Connecting sensor probes, swipe sample analysis instrumentation, and other sensors to such an infrastructure allows for instant data logging, analysis, forwarding, and review. It is important to note, however, that such improvements can not be freely selected without taking into account operator concerns and states' security and safety regulations. Many facilities will not allow wireless communication, for example, or the unsupervised taking of pictures by inspectors regardless of how important such capabilities and privileges might be for the success of a safeguards inspection. It is equally important to understand that such prohibitions are by no means an indication that the inspected Member State is trying to hide undeclared materials or activities; rather, it is following concerns about interference with, e.g., safety relevant instrumentation and confidentiality.

Where possible, however, such advantages should be applied towards safeguards. The use of hand-held devices can allow for the positioning and navigation through a site, and can integrate on-line services with a high security level. GIS-based display capabilities give inspectors the ability to access reports, maps, satellite images, points of interests, and any structured information stored at IAEA Headquarters from their mobile device. Inspectors can also validate or acquire new information and send it immediately to a central database in Vienna.

This technology can be implemented on a broad variety of hand-held devices enhanced by localization and communication add-ons like GPS, GSM, and satellite phone modules. Satellite links, GSM networks, or internet connections may be used for an online connection. If communication links are not permitted at the inspected facility, embedded data and downloaded applications still make this a useful tool during inspections. Where the use of GPS signals is possible and permitted, the instrumentation can support real-time location awareness of inspectors on site, adding the localization of materials, heavy equipment, and goods in the field as well as contributing to the safety and efficiency of the inspection.

GIS in AP applications reaches as far as the vision of real-time mapping to include not only the visualization and analysis of current location data, but also integrating real-time measurement data from sensor networks, location-related notes (requests, tasks, and orders), analysis results from fellow inspectors and from IAEA headquarters remotely accessed by the mapping platform.

## GIS and Mobile Applications

In future AP scenarios, the safeguarding of transported nuclear materials or sensitive technologies might be of interest. For certain nuclear security applications today this is already the case, and GIS-based solutions have proven to be an integral part of the tracking, monitoring, and alarm data analysis of nuclear materials transport. More precisely, the Global Threat Reduction Initiative (GTRI) that was initiated by the U.S. Department of Energy (DOE) is using proven safeguards technology concepts with GIS support.

GTRI's mission is to work "domestically and with other countries and international partners to identify, secure, recover, and/or facilitate the disposition of vulnerable, high-risk nuclear and radiological materials around the world."[3] GTRI aims at reducing global threats by consolidating such materials and returning them to their origin or other secure sites. For example, GTRI promotes the replacement

of HEU in research reactors with LEU and the return of the highly enriched materials to their origin. Naturally, this involves transport of nuclear or radiological material by truck, rail, ship, or even airplane, and the security and safety of such transports are of utmost importance.

In addition to providing safe and secure transport containers as well as protection of the transport, GTRI decided to implement additional, technical measures to ensure the integrity of the transported materials, using an adaptation of safeguards instrumentation together with a GIS-based management and alarm analysis tool. A surveillance system with cameras, balanced magnetic switches, seals, GPS, and communication infrastructure was developed to monitor the container while in transit, to ensure the integrity of the content, and provide alarm triggering in case unauthorized personnel approached the container or opened seals or doors.

All data gathered by the system, including images, seal data, time stamp, etc. are combined into a single data package. This data set (which can consist of alarm event or State-of-Health data) is then communicated through available channels (GPRS/EDGE, GSM, satellite, etc.) back to a surveillance center. The surveillance center has two functions: it displays alarm and State-of-Health data and images and it provides a GIS-based application to display all relevant data on a map. Figure 2 shows the basic operation of the GTRI system.

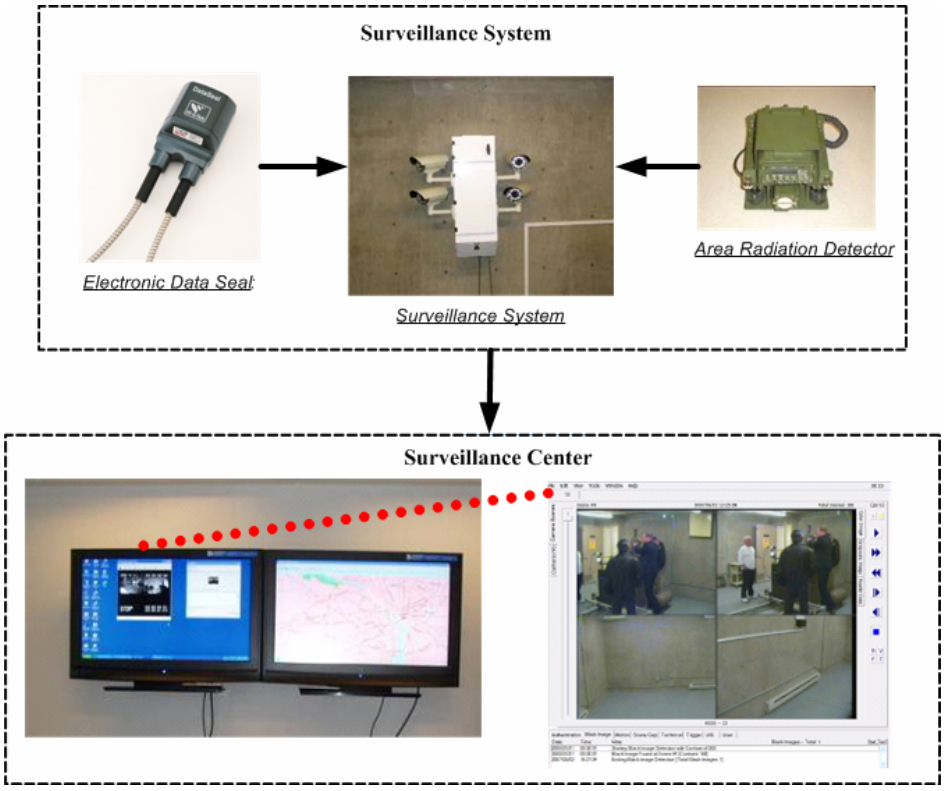


**Figure 2: GTRI Surveillance System**

The GIS-based mapping functions allow for the simultaneous display of all transports with shipments under the GTRI program. Monitoring officials can use this function to discern exact location and status of each container. If desired, operators can access near real-time images and other sensor data or just observe. In the event an alarm is signaled, appropriate response measures can be initiated. Figure 3 depicts a sample shipment tracking sequence.

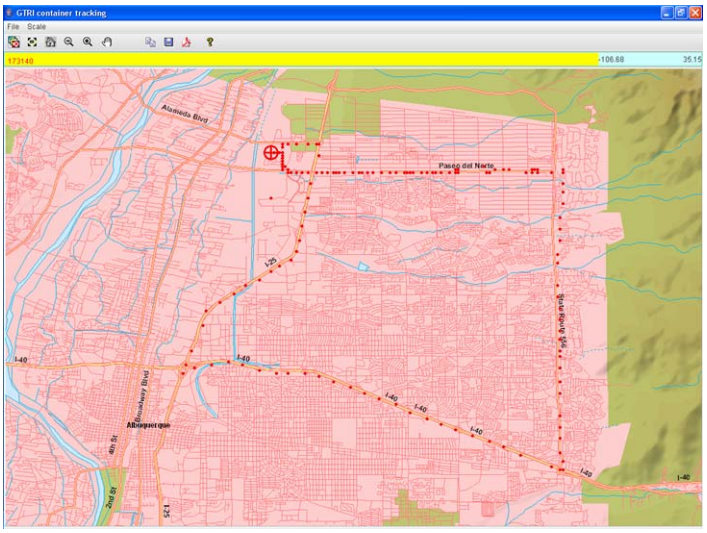


**Figure 3: Shipment Tracking Sequence**

## Information Management

As mentioned above, the AP indicates a shift in safeguards towards an information-driven approach rather than quantitative verification efforts. While this opens the door for a variety of interesting tools and activities for safeguards, it also poses the problem of how to cope with all the data and information. New analysis and detection methods are only of use if their information output can be analyzed in a coherent, effective, and efficient manner. This situation will become more and more challenging as the quantitative element declines and the qualitative element increases, always driven by the question of when the information acquired is sufficient to conclude with confidence on the absence of undeclared activities and materials.

GIS-based solutions can support AP safeguards efforts by allowing the logging and location tagging of a variety of data in a single position, consolidated at their place of origin. From there, other information analysis tools such as satellite imagery, wide area monitoring, nuclear forensics, open source analysis, or environmental sampling can integrate their output into a more holistic, geographical analysis. It will certainly not be possible to pack all relevant data into a mapping application, but it might serve as a place of origin consolidation tool that is easier to handle and query than multiple databases with limited or no inter-comparison tools.

## Conclusions

GIS-based solutions offer a variety of applications for safeguards, both in traditional and in Additional Protocol scenarios. The first steps of using data management and logging tools in mapping applications have already been initiated. While such solutions can support safeguards in stand-alone features such as in the NGSS application, the biggest advantage can be drawn from a holistic information management and analysis point of view. The need for such a comprehensive solution will increase as the importance of information analysis grows and as more and more techniques and technologies are added to the suite of AP measures.

The combination of multiple information sources and databases must be a joint effort and needs to be carefully managed both from an information use and a data access point of view. Also, such an infrastructure needs to be open to both data from new safeguards instrumentation and higher level information analysis disciplines to be a truly versatile tool for safeguards.

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# A Geo-Portal for data management and integration in Nuclear Safeguards

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***Abstract:***

*The implementation of strengthened safeguards measures creates a wealth of information including AP declarations, complementary access reports, inspectors' reports and Open Source information. Often, the information is scattered in different databases and document repositories. Therefore, efficient nuclear safeguards require an information platform integrating the collection, management and analysis of all relevant data. Since most safeguards relevant information has a geographical component (e.g. sites, buildings, mines), Geographical Information Systems (GIS) are suitable tools for satisfying the needs of a safeguards information system: the geographical features in the GeoDatabase are used to associate different data items in the distributed databases; the map-based interface allows to intuitively browse, query and find the information of interest, thus providing a single-point-of-entry.*

*Over the last years, the Joint Research Centre (JRC) has developed a number of tools supporting the preparation, management and verification of AP declarations, which were demonstrating the usefulness of Geographic Information Systems (GIS) for this purpose. Following these experiences, JRC in collaboration with DG-TREN developed SIT-ES, a Geo-Portal integrating the diverse safeguards-related databases at DG-TREN and providing a single point of access to all relevant information. SIT-ES allows DG-TREN users to have fast and direct access to information related to a particular facility or activity. The possibility to intuitively browse a map-based interface and quickly find the required information is essential for verifying the information and documentation efficiently and accurately. The paper will present the SIT-ES architecture, its main reporting capabilities in answering to clarification requests from IAEA or other interested parties, and will describe the main operational features:*

- *SIT-ES is a Geo-Portal, which provides an intuitive, map-based interface to access Safeguards related information through a geographic location on the map;*
- *It integrates different existing databases at DG-TREN and provides central access to the distributed information. Further to the Additional Protocol declarations, it links current and historic data for waste management, inspection planning, shipment of radioactive waste and spent fuel, decommissioning funds etc;*
- *It is a multi-user information system which ensures that all users always access the most up-to-date and accurate information;*
- *The system enforces the security requirements that come with this type of sensitive data. Each user can only access the data for which he has been authorised;*

**Keywords:** information management, data integration, GIS, Additional Protocol

## 1. Background

The goal of integrated safeguards is to improve the effectiveness and efficiency by integrating classical safeguards with the strengthened measures under the Additional Protocol (AP) while maintaining overall cost neutrality. The implementation of strengthened safeguards measures creates a wealth of information including amongst others AP declarations, complementary access reports, inspectors reports and Open Source information [1][2]. In order to achieve maximum efficiency, the responsible organisations require an integrated information system that allows the collection, management and analysis of all relevant data. Such a system must provide central collection and storage of different types of data, support data analysis, the generation of reports and the planning of on-site inspections. Many organisations have a series of legacy databases and data sources each with a separate interface that are not cross-linked which each other. A safeguards information system must integrate the existing databases and provide a single point of access to the user.

Most safeguards relevant information has a geographical component, i.e. it is linked to a geographical feature like a mine, site, facility or single building. Therefore, a Geographical Information System (GIS) is a suitable tool to satisfy the needs of a safeguards information system: a GeoDatabase contains all the geographical features of interest together with the cross-links to the relevant data items in the existing databases, thus integrating the diverse databases [3]. A GeoPortal provides a map-based interface in which the user can browse and query the GeoDatabase and which allows to link from each geographic feature to the relevant information in the existing databases, thus providing a single point of access for the user.

In the past, the Joint Research Centre of the European Commission (JRC) has developed a number of tools supporting the preparation, management and verification of AP declarations [4] showing the usefulness of GIS tools in the context of the Additional Protocol. Following these experiences, DG-TREN and the JRC are developing SIT-ES, a Geo-Portal which supports DG-TREN in fulfilling their obligations under the EURATOM treaty and the Additional Protocol. This article describes the objectives, features and architecture of the Geo-Portal.

## 2. Objectives

SIT-ES (Site Investigation Tool for European Safeguards) is the latest evolution of the SIT tools which have been developed to support the management, analysis and verification of AP declarations. It has been designed to the needs of the DG-TREN and is currently under development.

The Directorate General for Energy and Transport of the European Commission (DG-TREN) is responsible for the implementation of the EURATOM Safeguards system and the Additional Protocol within the EU member states. The objective of SIT-ES is to establish a single IT system that supports all parties at DG-TREN in performing the safeguards related tasks with the maximum efficiency and effectiveness. It will map the Nuclear Fuel Cycle infrastructures in Europe and allow the user to easily retrieve the information related to each of them. In particular it will support the Additional Protocol team and the EURATOM inspectors in their work.

The Additional Protocol team is responsible for managing and verifying the consistency of declarations submitted by nuclear sites operators in the EU member states. After verification, DG-TREN forwards the AP declarations to IAEA and, if necessary, answers any requests for clarification from IAEA. It collects and archives any useful complementary documents that can be used to better describe the sites. SIT-ES allows the AP team to have fast and direct access to their information related to a particular facility or activity declared under the Additional Protocol, including present and historic AP declarations, Complementary Access reports, inspection reports, Open Source documents, etc. The possibility to intuitively browse a map-based interface and quickly find the required information is essential for verifying the declarations efficiently and accurately. The reporting capabilities of SIT-ES help to answer clarification requests from IAEA or other interested parties.

The EURATOM inspectors are responsible for nuclear material accountancy verification in member states facilities and are involved in complementary access visits under the Additional Protocol. Fast and easy access to all relevant information is an important asset to their work. For example, the geographic features of SIT-ES, including the site drawings with the related information of each building can contribute to the planning of site inspections.

## 3. Features

In order to achieve the objectives described above, SIT-ES has been designed as a multi-user GIS system with a central geo-database and a Geo-Portal as interface to the end-user:

- It is a multi-user information system: expert users are responsible for validating, inserting and maintaining the data. End users then browse the data for read-only access. This ensures that all users always access the most up-to-date and accurate information. The system assigns different roles to users according to their function: expert users (i.e. members of the AP team) have full read/write access to the data; end users have read-only access to all or parts of the data.
- It is a Geographic Information System (GIS): a GeoDatabase contains all geographic features which are of interest for European safeguards, for example mines, research locations and nuclear sites including the buildings declared under the Additional Protocol. The interface allows intuitively browsing maps at different scales (from state level down to the site level), viewing the content of the GeoDatabase in its geographical context and retrieve the associated information.
- It integrates the different legacy databases at DG-TREN (containing for example AP declarations, material accountancy, complementary access reports) and thus acts as a Geo-Portal to the existing information: for each entry, the GeoDatabase contains the cross-links to all related information items in the existing databases (see Figure 1). For example, for each nuclear site it contains a link to all AP declarations related to the site, all complementary access reports, all Open Source information, etc. On the SIT-ES map interface, the user can then browse to the site and request any related information available in the system with a simple mouse-click. The request is forwarded to the application that handles the specific database, which then takes care of the retrieval and visualization of the data (see Figure 2).

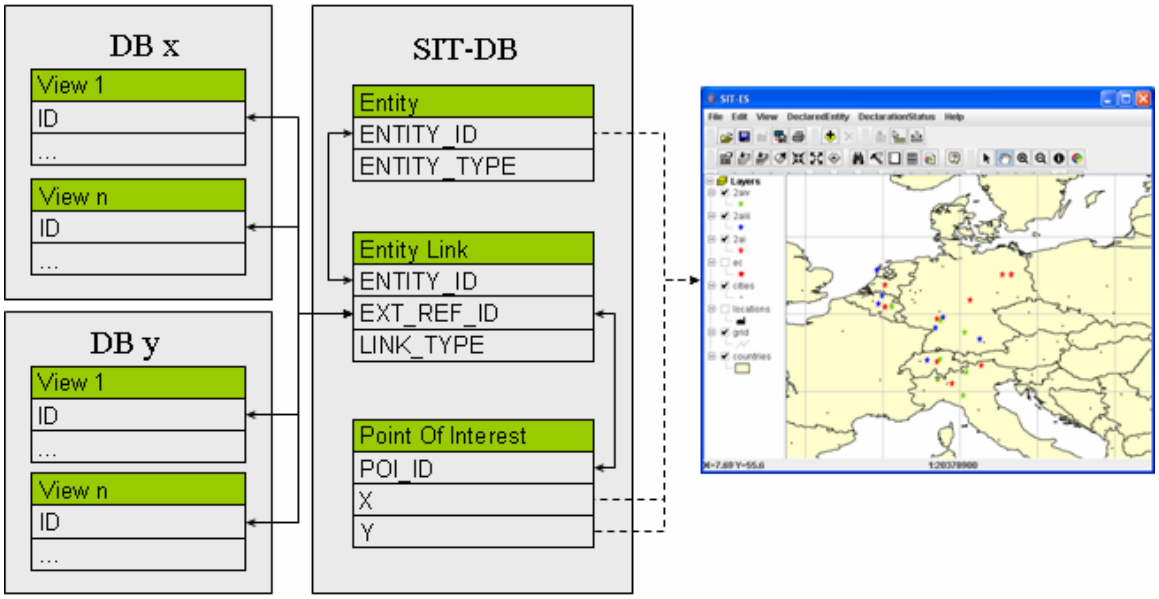


*Figure 1: Simplified structure of the SIT geo-database(SIT-DB) showing the links to the external databases. The geographical client can retrieve information about the entities, their geographical coordinates and basic information provided through the external database views.*

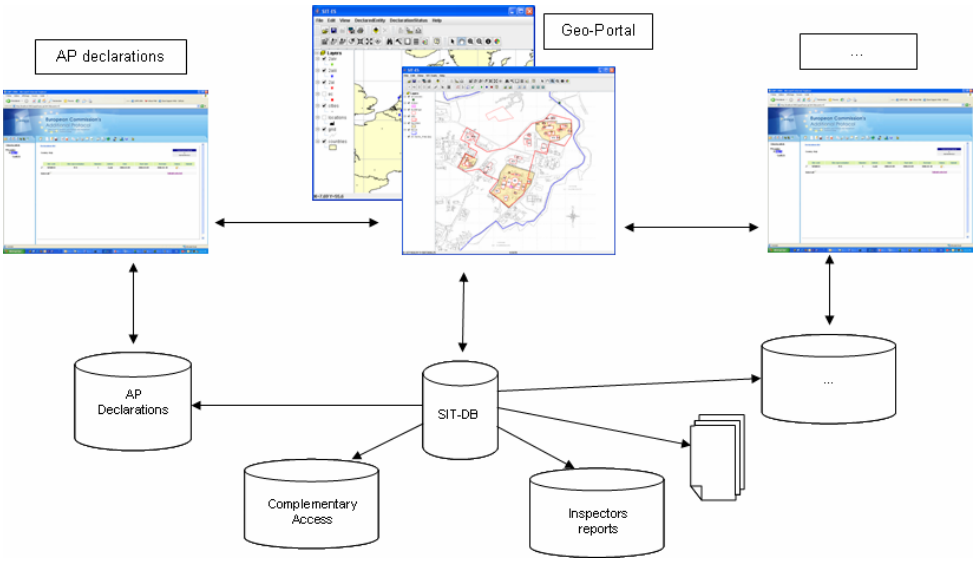


*Figure 2: The map-based interface of the Geo-Portal is the entry-point for the user. He can browse and select the feature of interest and query it for further detailed information, which will be provided by the existing database applications.*

- The system enforces the security requirements that come with this type of sensitive data. It is installed within the DG-TREN network, where strict security measures ensure the integrity of the data. SIT-ES does not retrieve any of the sensitive business data in the existing databases directly, but provides the user a link to the specialized applications which takes care of the data retrieval and associated security issues Furthermore, a role-based access model ensures that each user is provided only with the functionality for which he is authorized.

Figure 3 shows how overview information can be displayed directly on the SIT-ES interface and Figure 4 illustrates how external applications are launched to view detail information on a specific geographic feature.

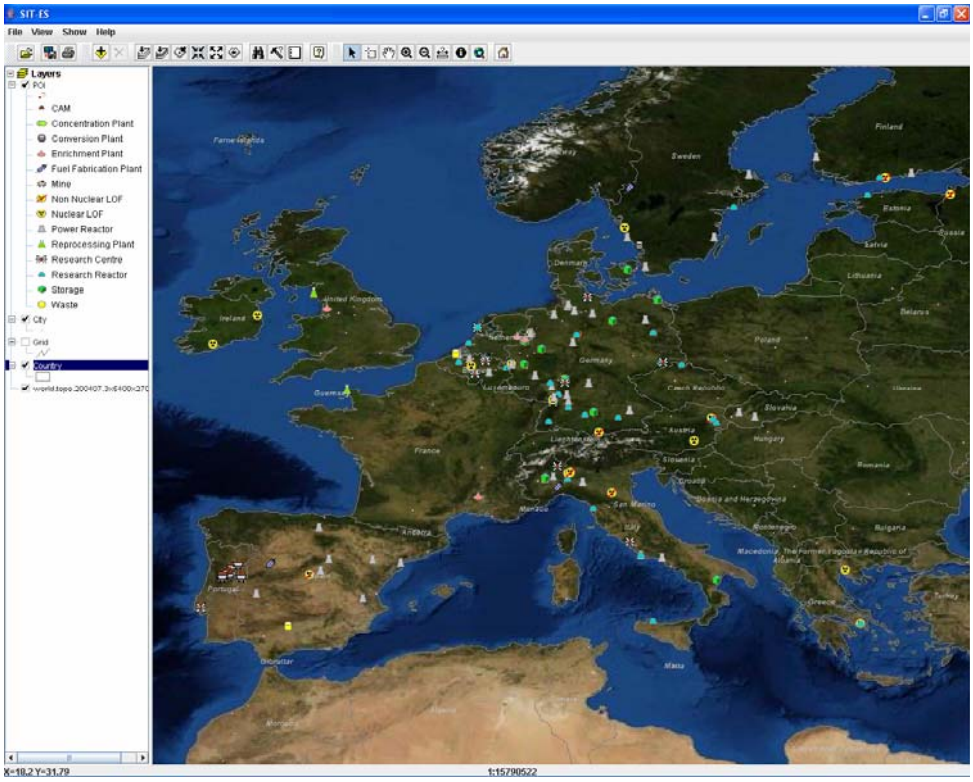


*Figure 3: Overview map showing the SIT entities classified by installation type.*

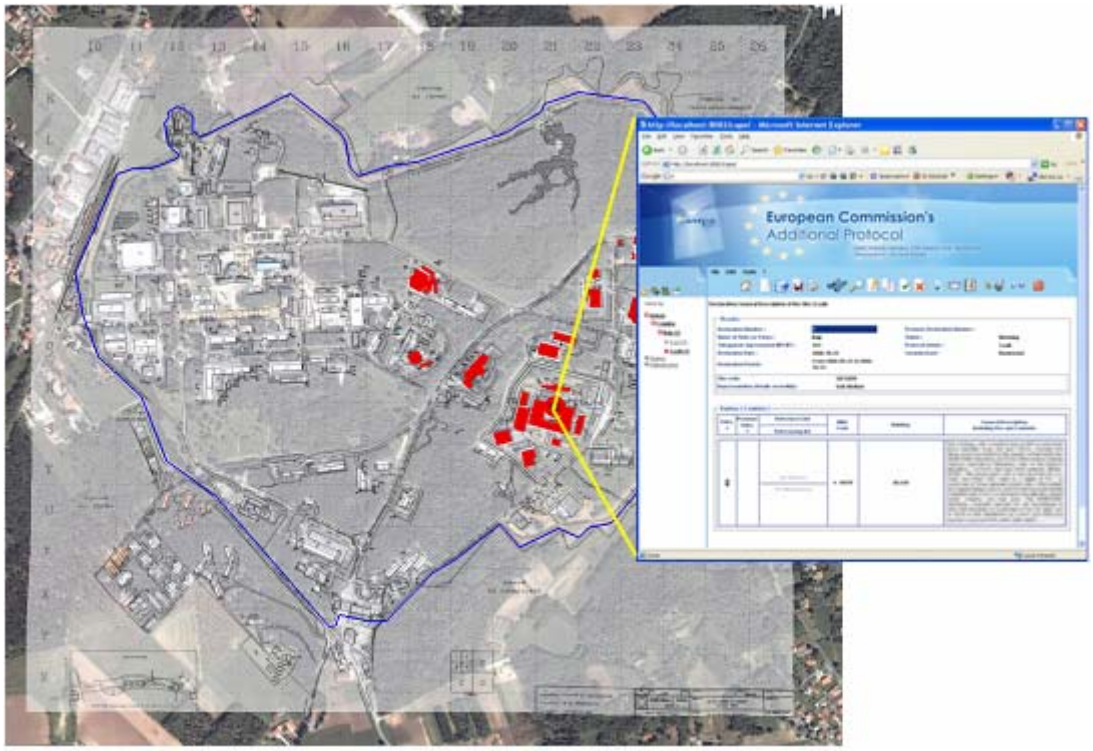


*Figure 4: Geographic site information (satellite image, site layout and building vectors). The AP application is launched for a specific building through a single click on the vector representation.*

## 4. Conclusion

The development of the Site Investigation Tools (SIT) has shown that Geographic Information Systems (GIS) are suitable tools for the management and analysis of data related to the Additional Protocol. They put the safeguards related information in their geographical context and the GIS interface allows the user to quickly find and retrieve information by querying the associated object (e.g. a site or a building) on a map.

The JRC and DG-TREN are currently developing SIT-ES, a multi-user GIS system that integrates existing safeguards-related databases at DG-TREN and acts as a Geo-Portal to all safeguards-related data available at SIT-ES. The system will support the DG-TREN in fulfilling their obligations under the EURATOM treaty and the Additional Protocol with maximum efficiency and accuracy.

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ESARDA 29th Annual Meeting, Symposium on Safeguards and Nuclear Material Management, Aix-en-Provence (France), May 22-24, 2007

# About Ground Penetrating Radar

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# Presentation outline

- Ground Penetrating Radar and DIV in NPT Additional Protocol
- Ground Penetrating Radar general presentation
- Implementation of Ground Penetrating Radar in IAEA Inspection
- Other geophysical methods to combine to Ground Penetrating Radar
- Conclusion

# Ground Penetrating Radar and DIV in Additional Protocol

- **IAEA Safeguard context**
  - Safeguard objectives : Ensure absence of undeclared activities & facilities
  - NDA such as geophysics is an integral part of the verification activities
  - Additional Protocol includes new plant design, New plant specific NDA, increased performance for existing NDA, novel tools to meet new objectives

- **Geophysics in DIV**
  - Geophysics methods could be of interest for containment and Design Information Verification to strengthen safeguard capacities
  - IAEA has identified ground penetrating radar as a potential novel method that could improve inspection by non-invasively mapping
  - Other potential useful geophysics methods to be investigated
  - Main objective : Detect undeclared facilities (doors, tunnels, storage …)

- **Ground Penetrating Radar for DIV in IAEA context**
  - Developing procedure, determining capacities, developing dedicated training

- **Task do not cover use for deep repositories verification nor aging NDA**

# Presentation outline

- Ground Penetrating radar and DIV in Additional Protocol
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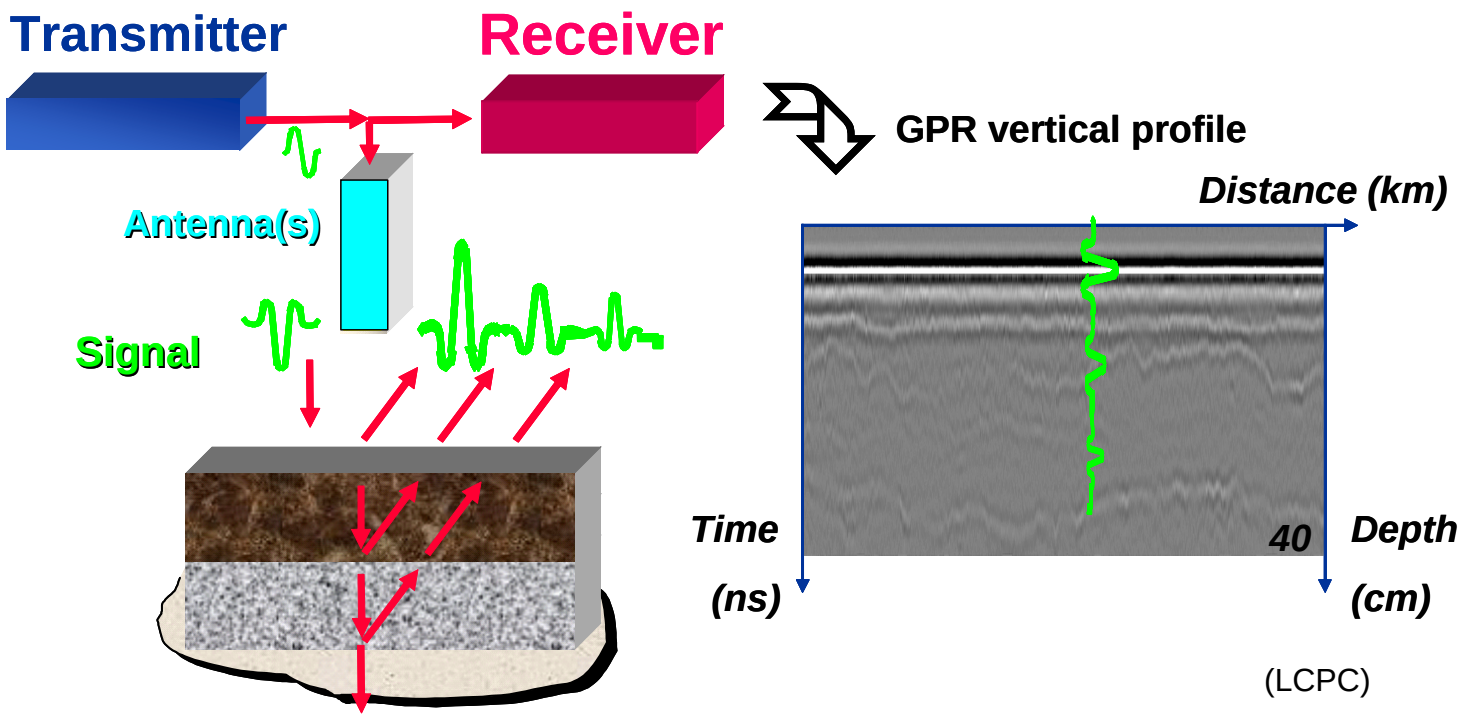
- Conclusion

# Ground Penetrating Radar general presentation



- GPR uses HF pulsed electromagnetic waves to map subsurface information.
- Transmitting and receiving antennae are dragged along the ground surface
- Transmitting antenna radiates short pulses of HF radio waves into the ground
- The wave spreads out and travels downward. Waves reflected on objects or boundary with electrical properties contrast are recorded by receiving antenna
- The principles involved are similar to optic or reflection seismology



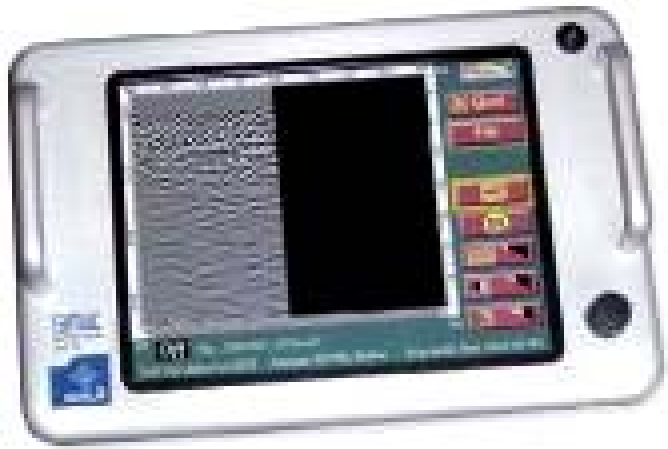
# Ground Penetrating Radar general presentation



Some GPR devices with Agency tools
antennae ranging from 50 MHz to 2 GHz

**(Courtesy Sensors&Software)**

**(Courtesy GSSI)**



**(Courtesy Malå)**

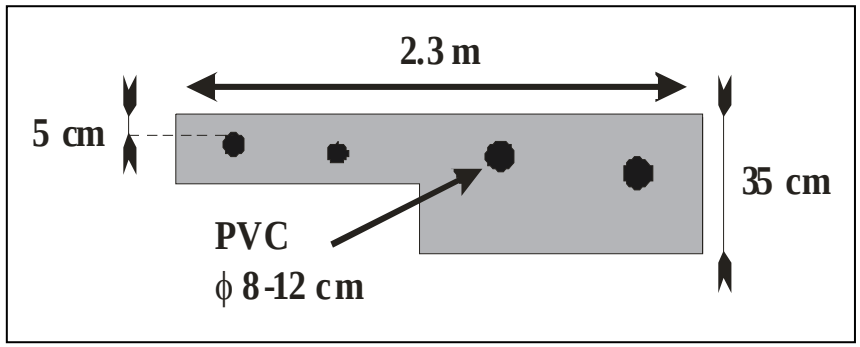
**(Courtesy Sensors &Software)**

**(Courtesy GSSI)**

# Ground Penetrating Radar general presentation

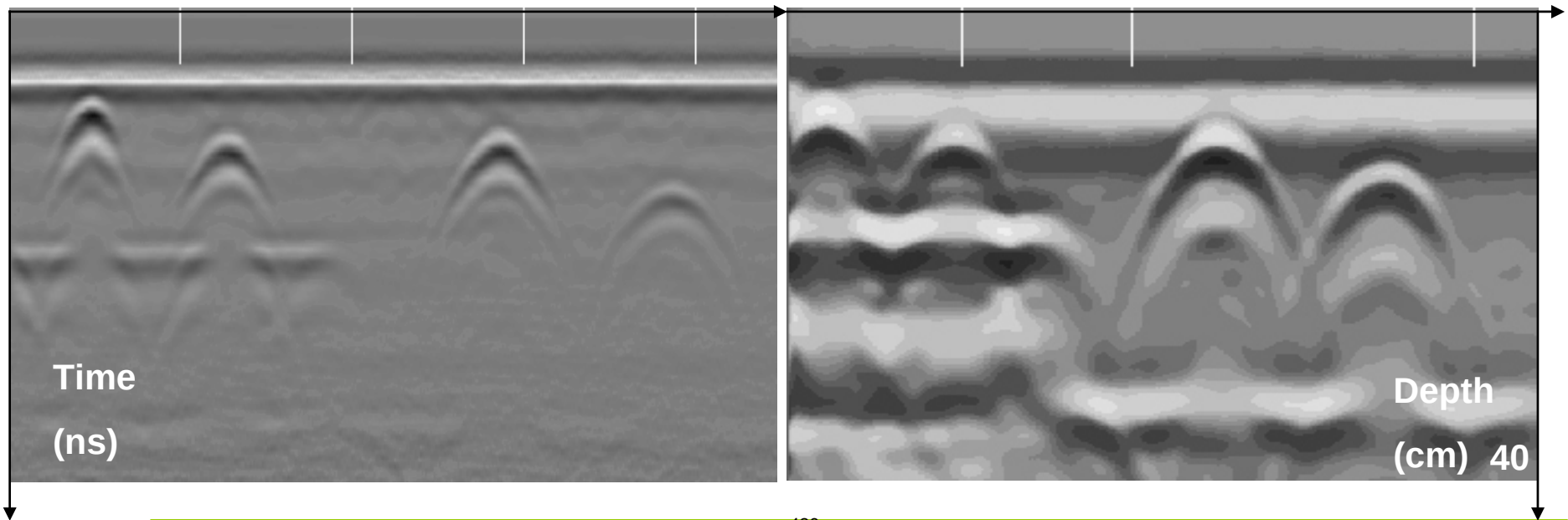


Penetration (few cm to nx10m) is driven by exponential attenuation linked to frequency emission, medium conductivity, surface nature



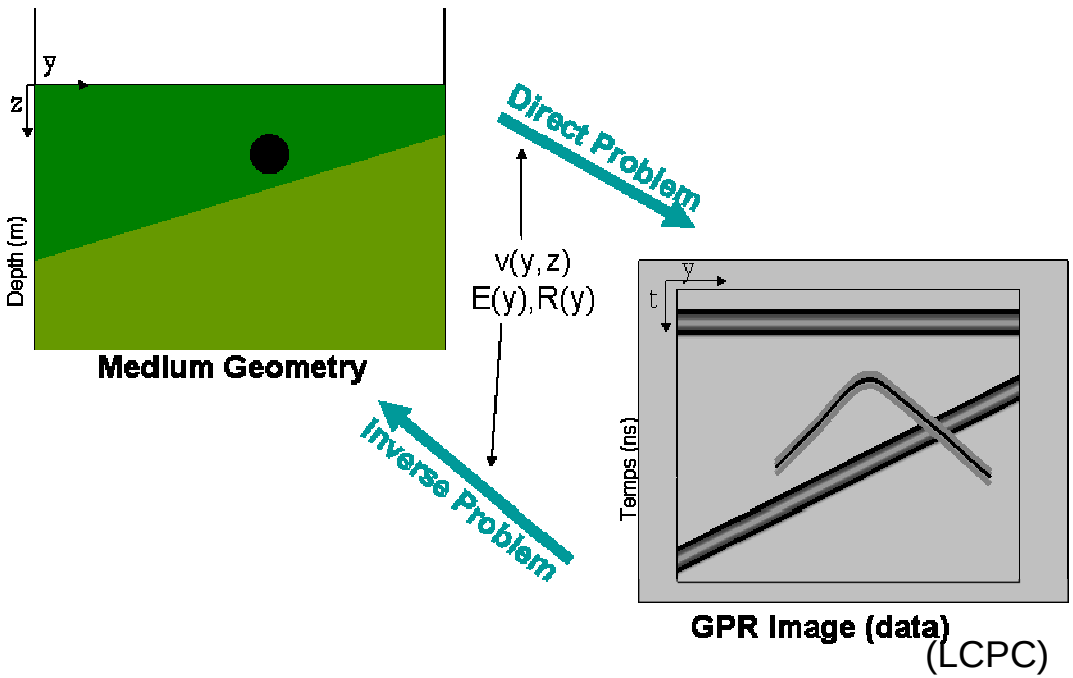

(1.5 GHz antenna)

(900 MHz antenna)

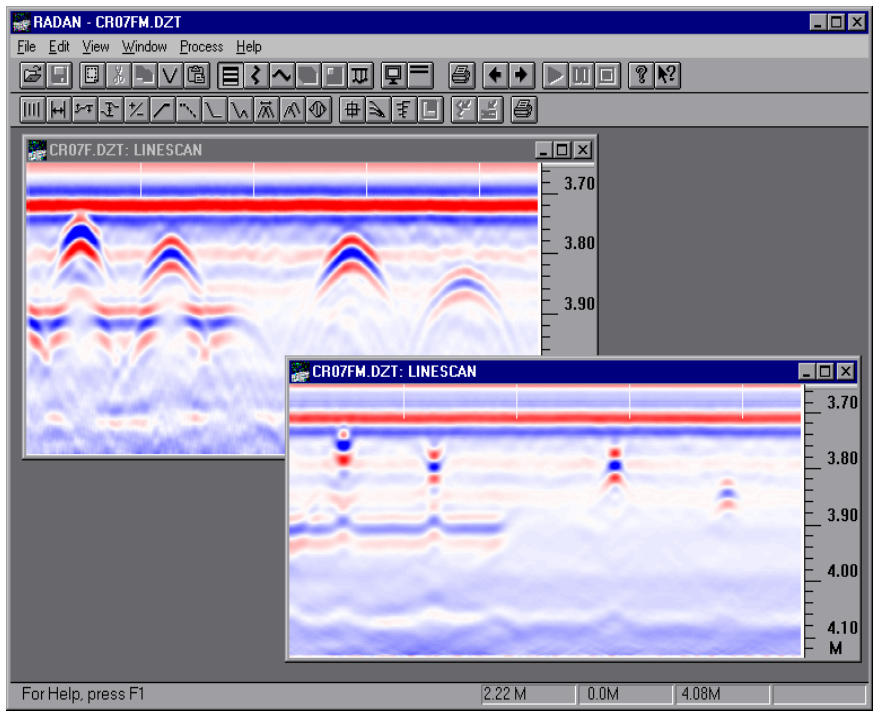


# Ground Penetrating Radar general presentation

- Basic fast Processing of recorded raw data :
  - Amplitude decay compensation
  - Answer to the inverse problem for real geometry access
  - Time to depth conversion with average velocity
  - Object parameters identification




(LCPC)



# Ground Penetrating Radar general presentation



- GPR general capacities
  - Light and portable, user friendly
  - Field tested and proven
  - Real time processing and (quite) real time display
  - Contact not required
  - Object nature identification (f.i. voids)
  - May detect metal as well as plastic objects
  - May image though a concrete slab or wall

- GPR general limitations
  - **Physics** : Multiple, lateral reflections, attenuation, indirect image of structure
  - **Instrumentation**: Frequency of antennae, form and duration of the pulse
  - **System-dependent effects** : oscillations within the signal, “ringing”, real time processing and visualization, link with localisation system
  - **Site-dependent effects**: sensitive to EM noise, medium conductivity (f.i. clay-rich zones), surface roughness, sensitive to water

  *… limitation factors that might be useful information when locally abnormal*

… Capacities to be analysed for IAEA targets and operations context

# Presentation outline

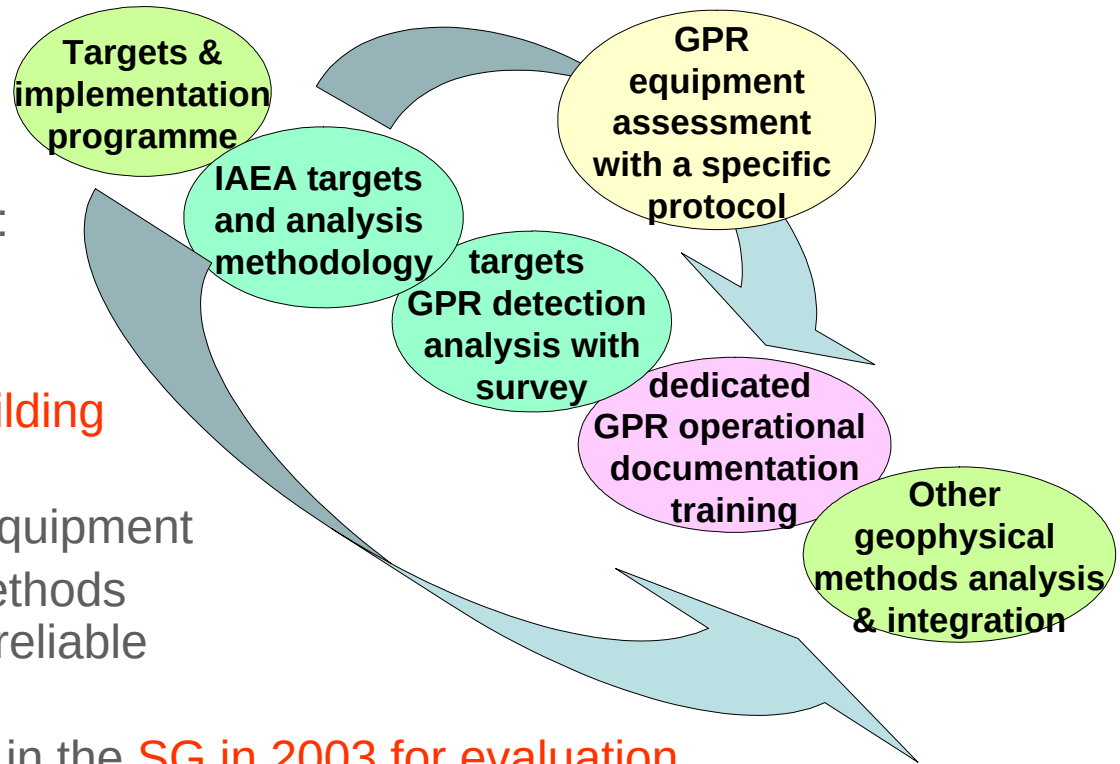
- Ground Penetrating radar and DIV in Additional Protocol
- Ground Penetrating Radar general presentation
- **Implementation of Ground Penetrating Radar in IAEA Inspection**
- Other geophysical methods to combine to Ground Penetrating Radar
- Conclusion

# Implementation of GPR in IAEA Inspections / programme

Objective : verify the completeness and correctness of DIV declaration

- Toward Agency suited GPR implementation

- Necessity to identify capacities, limits and appropriate usage for selected situations (Global Configuration) to investigate :
  - Structures inside a wall or slab
  - Structures behind a wall or slab
  - Structures in the ground near building




- Need for an evaluation of available equipment
- Identification of other geophysical methods capacities to get interpretation more reliable

- Ground penetration radar introduced in the SG in 2003 for evaluation

- 3Y programme conducted by France as MSSP c/o CEA with LCPC & CGG
- Based on French AGAP geophysical quality guides

# Implementation of GPR in IAEA Inspections / programme



- Guides for inspection preparation by IAEA head quarters

**To be verified detected identified**

A kind of structure
A void behind a wall, …

*Elementary Questions*

**Possible structural signatures and context analysis**

Nature, thickness of material

*Typical Requirements*

**Preparation of measurement plan**

Methods selection (GPR, IR, …) and survey configuration

*Application guide & ref. manual*

- Implementation of the DIV with “on site” inspection plan & survey

**On site implementation reference procedures**

Device implementation procedure

Reference picture expected

*Reference Manual*

**On field QC & data assessment**

**Backstage advanced data processing and interpretation**

Data quality control

First level interpretion

*Reference manual*

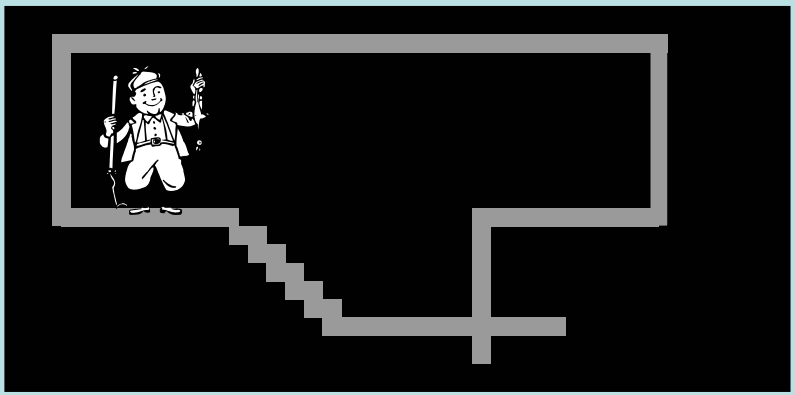
# Implementation of GPR in IAEA Inspections / analysis



- GPR capacity analysis : theoretical & generic approach based on situations analysis

## 1. Global Configuration - IAEA

**(Case histories - Environment)**



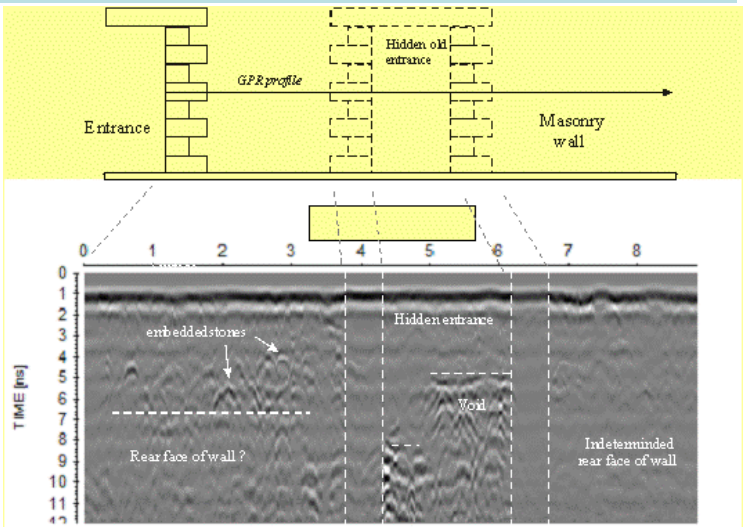
## 2. Elementary Questions/ GC

**Q1 : Characteristics of the wall or slab**

**Q2 : Filling under the slab**

**Q3 : Filling behind the wall**

**Q4 : Structure extension**




## 3. Typical Requirements common to all global configurations

**A1 – Thickness of a wall or a slab**

**A2 – Identification of the [various] constitutive material of a wall or a slab [f.i. filled door]**

**.....**

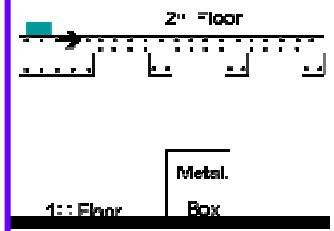
→ Case studies & modelling

→ Appropriate techniques

→ GPR survey configuration

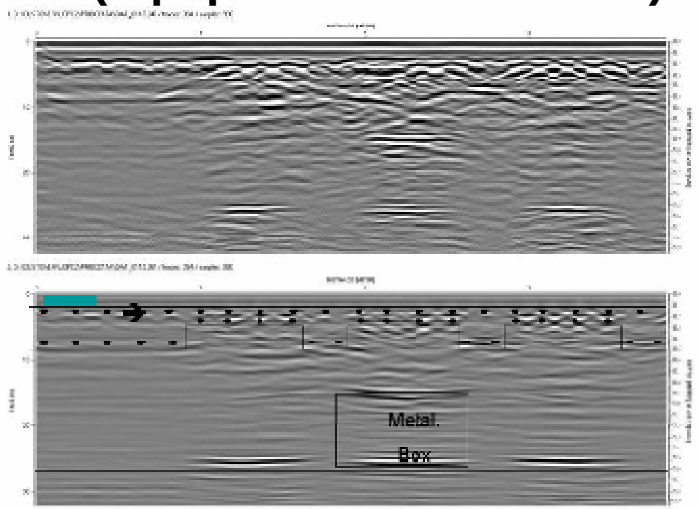
# Implementation of GPR in IAEA Inspections / analysis

- GPR method capacity analysis : other examples

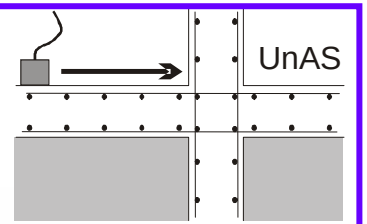


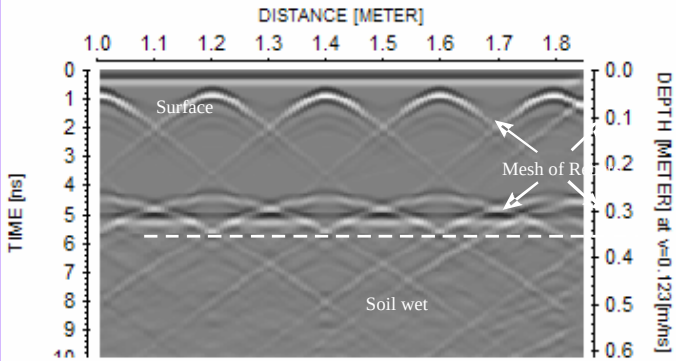

Raw & processed GPR profiles (800 MHz):

**(equipment below a slab)**



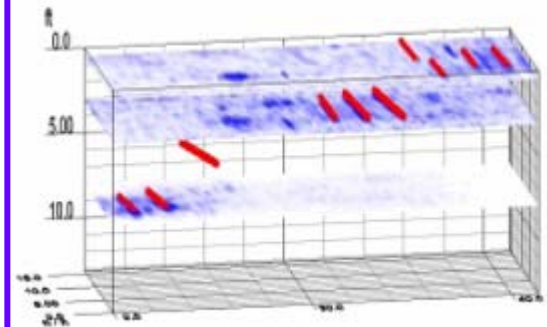

**(buried room & wall)**

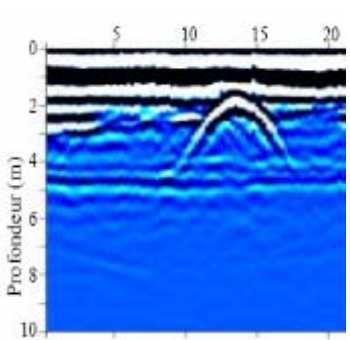


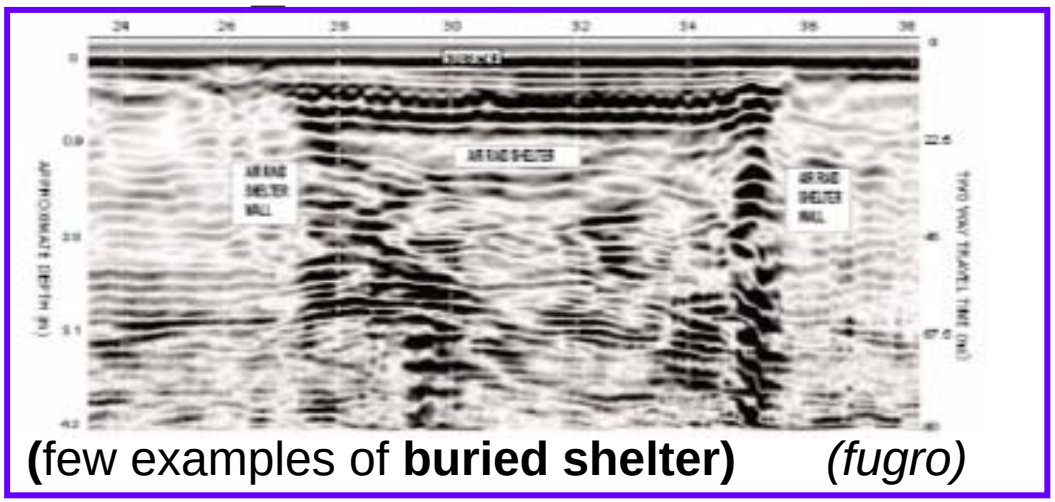

*(lcpc)*

**(buried pipes**






*(gssi)*



**(**few examples of **buried shelter)** *(fugro)*

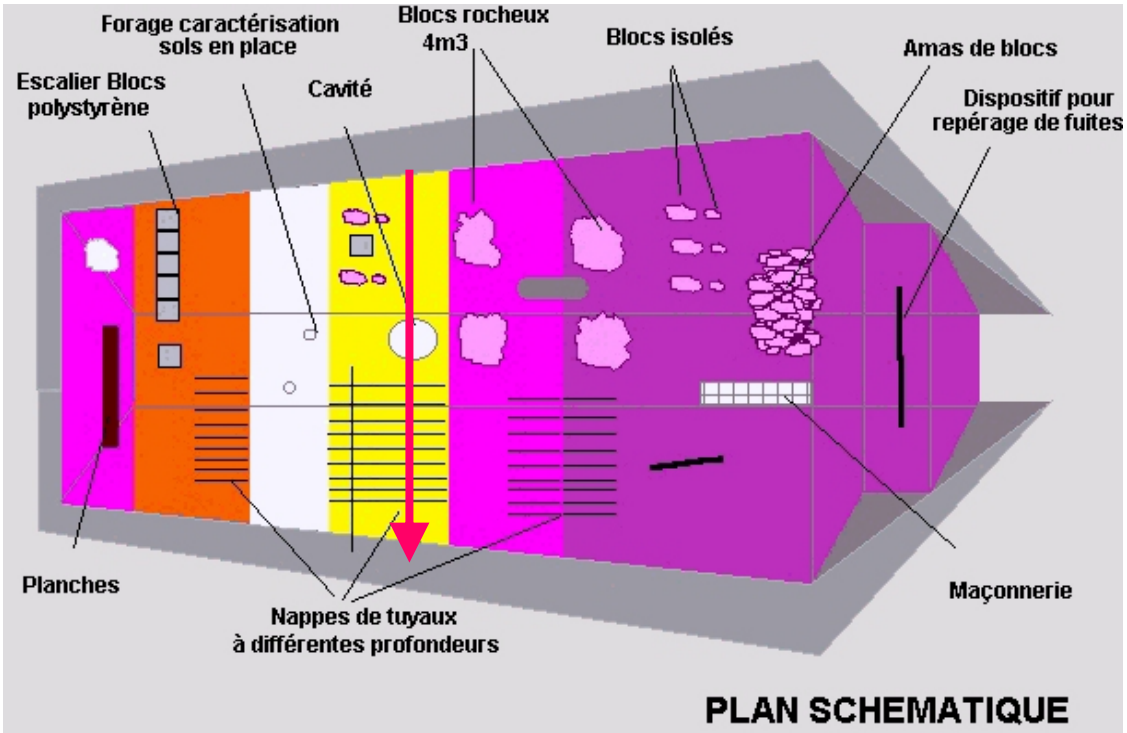
# Implementation of GPR in IAEA Inspections / Equipment test



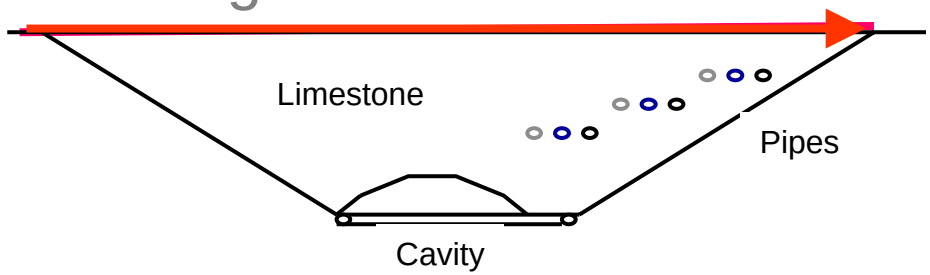
- **Market equipment dedicated assessment**
  - **Four market equipment manufacturers with various antennae**
  - **Assessment with fixed protocol including processing**
  - **Four main applications classes :**
    - deep geophysics
    - shallow geophysics
    - global structure non destructive assessment
    - local structure non destructive assessment
  - **Analysed parameters**
    - **Workability** : system, cart, cables, time duration …
    - **Implementation**: setup, recorded data, preprocessing, …
    - **Radar performances**: Frequency bandwidth, Noise level, Penetration
    - **Price**: elements, cables, ….

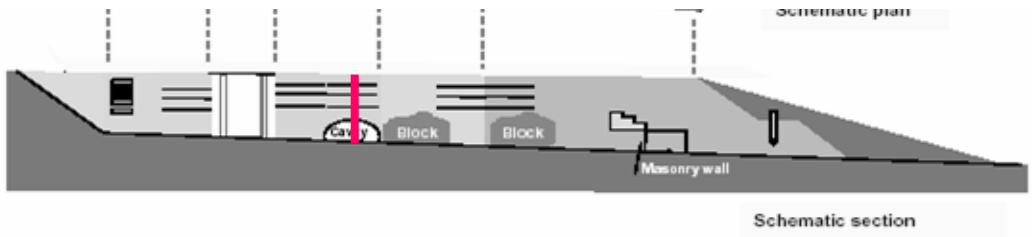
# Implementation of GPR in IAEA Inspections / Equipment test

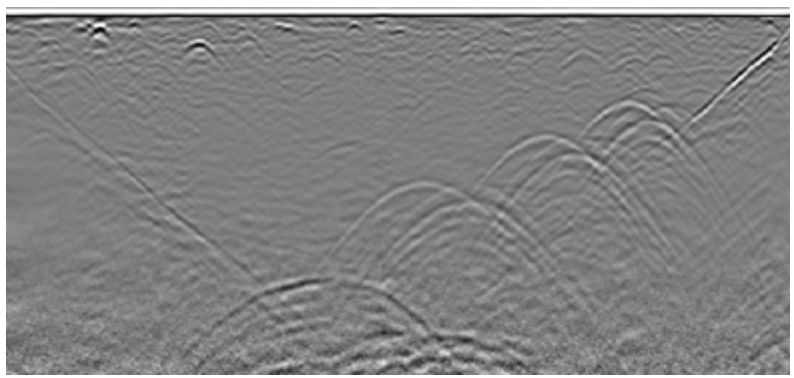





## Equipment assessment using LCPC test site



# Implementation of GPR in IAEA Inspections / Equipment test



2 weeks with systems test in LCPC :
*Courtesy of GSSI, Mala, IDS, RSI*

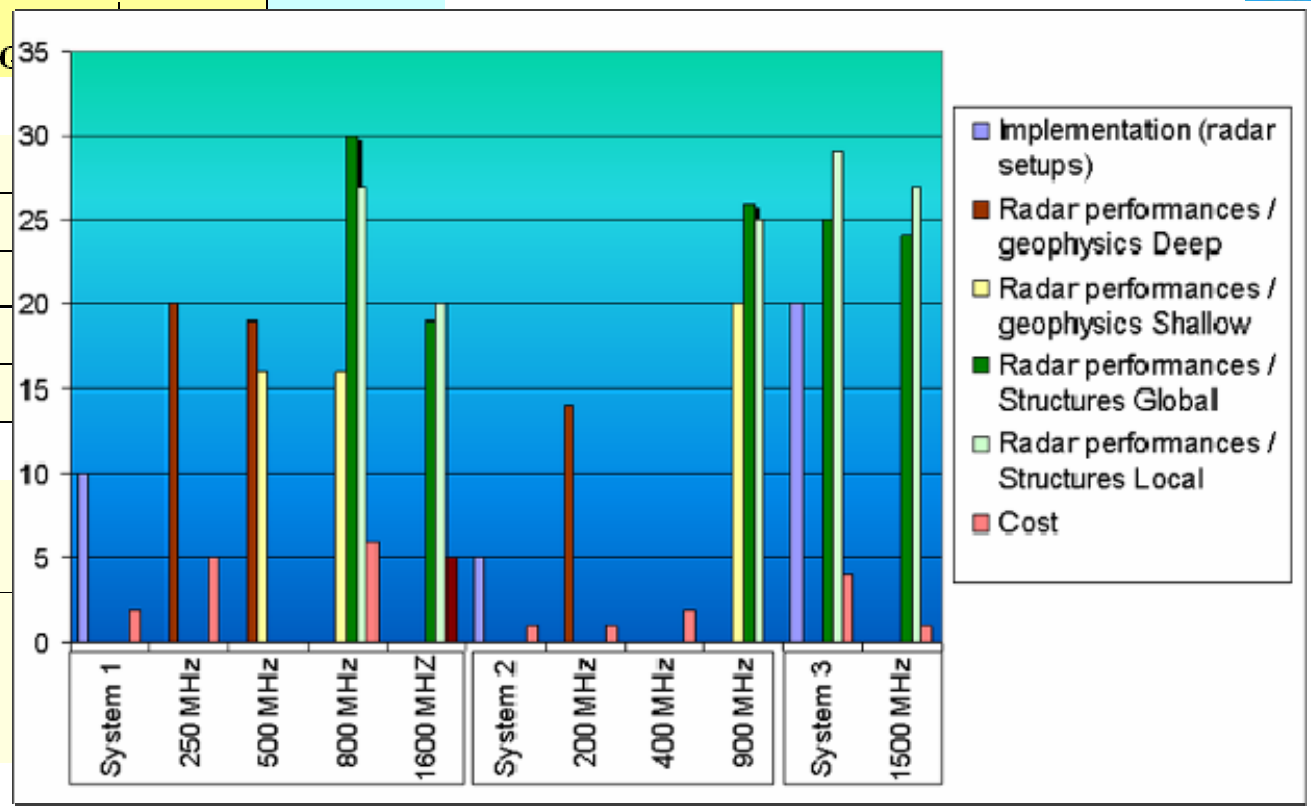


# Implementation of GPR in IAEA Inspections / Equipment test



## Equipment assessment global output

| | Implementation (radar setups) | Radar performances / geophysics | | Radar performances / Structures | Prices[a] |
|---|---|---|---|---|---|
| | | Deep | Shallow | G | |
| **System 3** | | | | | |
| - | 4 | | | | |
| **200 MHz** | | 20.5 | - | | |
| **600 MHz** | | 24.5 | 23 | | |
| **900 MHz** | | - | 19 | | |
| **1600 MHz** | | - | - | | |
| **System 4** | | | | | |
| - | 8 | | | | |
| **300 MHz** | | 16.5 | - | | |
| **500 MHz** | | 23 | 19 | | |
| **900 MHz** | | - | 27 | | |
| **1500 MHz** | | - | - | | |



# Specific and dedicated inspectors training 2006'nov.

- ✓ **Specific and dedicated** documentation as well as training programme
- ✓ Include theory and also tasks **from acquisition to basic processing**
- ✓ Covers inspection preparation and field measurement
- ✓ Basis for **quality control and data quality assessment**
- ✓ Focus on appropriate survey parameters & equipment configuration for Agency targets
- ✓ Includes **site observation guide** and **data relative localisation**
- ✓ Organized on LCPC site with Agency training section
- ✓ **Equipment** : IAEA (Mala & GSSI), LCPC (GSSI) and rent (Mala)
- ✓ Same ReflexW processing, modelling and interpretation **software**

# Presentation outline

- Ground Penetrating radar and DIV in Additional Protocol
- Ground Penetrating Radar general presentation
- Implementation of Ground Penetrating Radar in IAEA Inspection
- **Geophysical methods to combine to Ground Penetrating Radar**
- Conclusion

# Geophysical methods to combine to Ground Penetrating Radar

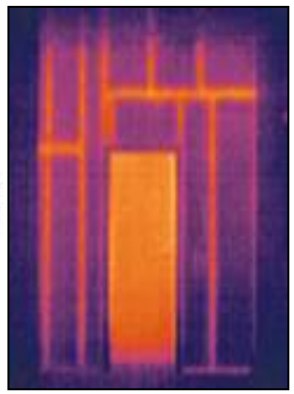


- Identification of methods providing complementary information related to other physics (acoustic velocity, density, magnetism, …) to confirm nature

**(Ultra Sonic Pulse Echo)**

**(Impact echoe)**

**(Thermal imagery)**



**(Magnetism)**

**(Electromagnetism)**

**(& MASW, …**

- First step excluding heavy, slow or complex methods (ex : gravimeter)
- Identification and begin of analysis of appropriate of a first set of field proven and light complementary methods, to be merged in data fusion process

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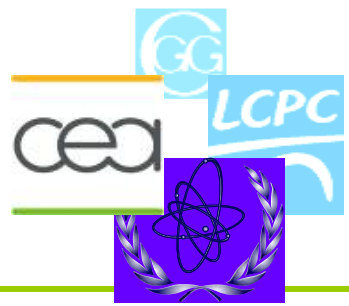
# Conclusion

- Conducted dedicated programme, documentation and training should allow **useful measurement** on the field, by using GPR when appropriate and by **providing useful good data**

- Support programme helped Agency to identify capacities and to reduce GPR limits for a first set of situations and targets
  - Management and reduction of method limits by specific analysis of identified targets
  - Limitation of equipment reduced by equipment knowledge provided by systems tests
  - Limits related to required skills reduced by a dedicated training
  - Trend during inspections and better idea of capacities and appropriate survey parameters & procedures
    - For building DIV with better capacities from now
    - For buried structures DIV, more complex, other techniques to be combined to ensure results

- Trends for continuation … in a continuous improvement process
  - Training minor improvement and continuous training development
  - Enlarge target spectrum, adjust LCPC test site, link with other Member state programmes
  - Implementation of other techniques should be useful to improve inspection and to cover larger spectrum
  - Generic methodology should be useful for other techniques or target analysis, with data fusion to handle

- **Present programme results linked to strong and efficient collaboration and complementary skills of Agency, LCPC, CGG and CEA/DASE**



# Nuclear Fuel Cycle Facility Monitoring

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***Abstract:***

*This paper presents the result of several projects dealing with nuclear fuel cycle monitoring. These projects have been developed for more than ten years and reveal the relevance of this approach of local change detection (local event) in periodic curves or signal (cycle).*

*After a presentation of the syntactical pattern matching used, a description of the algorithm capabilities in pattern recognition and anomaly detection are given.*

*A first application is described. It is focused on level/density/temperature recognition for vessel operation in La Hague reprocessing plant (in the framework of Euratom Treaty). The purpose is to detect discrepancies between operator declarations and a near real-time accountancy based on solution transfers. This method has been generalized to other periodical signal such as industrial weighing devices.*

*In the mean time a second application of the pattern matching principle has been implemented for Rokkasho Reprocessing Plant (in the framework of Non Proliferation Treaty). It has been done with some specific differences in local event (Standard Event) interpretation.*

*Solution monitoring will not work without a good measurement of volume an density. Regarding volume this paper presents bubbling system description as the one being delivered to the Mayak RT-1 Reprocessing Plant in the framework of the TACIS program.*

*Finally, the pattern matching concept is also presented for a non destructive assay monitoring application. It allows detecting regions of interest for spectrum analysis algorithm, following nuclear material along production line and checking for the correctness of operator declarations.*

*These three applications are also described in terms of software architecture and technologies, showing therefore the generality of this approach.*

*As a conclusion a description of a global and vertical monitoring system for nuclear solution monitoring is given.*

**Keywords:** Monitoring; Pattern-matching; Reprocessing

## 1. Introduction

The ability of a safeguards system to provide additional assurances on operator declarations is essential for Safeguards Inspections. A help can be obtained using an Information Technology (IT) system able to monitor plant operations and to implement the Inspection and Verification tools. The detection of discrepancies between the declared material balance and Inspectors' computed balance can therefore be seen as a diagnosis problem on huge and complex plants. If this detection is near real-time it allows timely detection of anomalies. Near real-time monitoring enabled by IT systems is a significant improvement over waiting for annual inventory periods to make these detections.

Nuclear material accountancy is of primary importance for nuclear Safeguards. Within a reprocessing facility, the chemical processing involves large volumes of nuclear material solution in numerous vessels. Discrepancies in material volumes (and mass) transferred between a feeding and a receiving tank may be of great importance for nuclear material accountancy. Solution monitoring is an ideal tool for safeguarding the chemical processes in a reprocessing facility.

IT architectures for solution monitoring merge constraints from Safeguards, diagnosis and technical viewpoints. For instance, confidence in a solution monitoring system depends on avoiding false acceptance and false alarms. There are several diagnostic methods for detection of discrepancies between actual data and a model. Each method has drawbacks and advantages in terms of modeling effort and error propagation. Finally technical considerations influence the system.

Key constraints for solution monitoring are classified below:

1. Safeguard constraints:
    - Inspector alarm review tool must give contextual information to help understand of the situation.
    - The underlying infrastructure must be transparent.
    - Inspector resources are limited therefore the monitoring system itself must be as automated as possible to prepare results.

2. Diagnosis constraints :
    - The system must consider all transfer piping arrangement for transfers.
    - The system must be accurate regarding volume, mass and time variables.
    - The system must be easy to configure and maintain.
    - The system must be fast enough to keep up with a continuously operating facility
    - The system must synthesize all data into knowledge.

3. Technical constraints :
    - Graphical User Interface (GUI) must be compliant with modern standards.
    - The system must ensure data integrity: no relevant data can be lost or modified by a third party.
    - The system must ensure drill down capabilities from high level data structures to raw data.
    - The system must allow the fast storage of time-stamped data and of 'event' structures.
    - The system must give information about internal state-of-health.
    - The system must allow self monitoring regarding power supply.
    - The system must centralize all information in a unique system (the monitoring reference) through one or two networks.

The reader who is interested by a complete list of recommendations and standards can refer to [1] which encompasses the family of constraints for both DG-TREN and IAEA viewpoints in the scope of future monitoring and non destructive assay (NDA) measurement systems.

This paper presents technologies and concepts for monitoring material solutions in nuclear reprocessing facilities under the framework of verification regime for international treaties. This paper addresses the following systems:

- Solution Monitoring of AREVA NC La Hague reprocessing plant in the framework of the Euratom Treaty (SYSTEM7), with some highlights on other monitoring extensions.
- Solution Monitoring of Rokkasho reprocessing plant (RRP) in the framework of Non-Proliferation Treaty (NPT).
- The NDA Review (NDAR) at RRP in the framework of NPT
- The bubbling system developed for the MAYAK RT1 Reprocessing Plant the Russian Federation, in the framework of the TACIS Program.

These activities are industrial projects; the older one has been running for more than ten years [2].

## 2. Outline

It is useful to underline three majors aspects of solution monitoring for the detection of material balance discrepancies. These are: the process complexity itself, the system diagnostics and the data.

### 2.1. Process complexity

The balance equations to be solved must be dynamically connected to the selected piping arrangement. In order to detect material balance discrepancies with operator declarations, the follow up of nuclear solution transfers requires accurate and synchronized time data since transfer's start and end time are the integration boundaries for mass computation.

The balance equations become quite complicated for interconnected vessels. In particular, vessels having batch input with continous output or continuous input with batch output or batch inputs and outputs such as accountancy vessels.

Moreover some vessels can have several feeding and receiving tanks. The figure 1 below provides an exemple of an accountancy vessel and connected tanks. The solution monitoring must provide a way to connect mass balance equations around the accountancy vessel, following the actual and current tranfer chosen by the operator (for instance with feeding from, one or two vessels and receiving in only one).

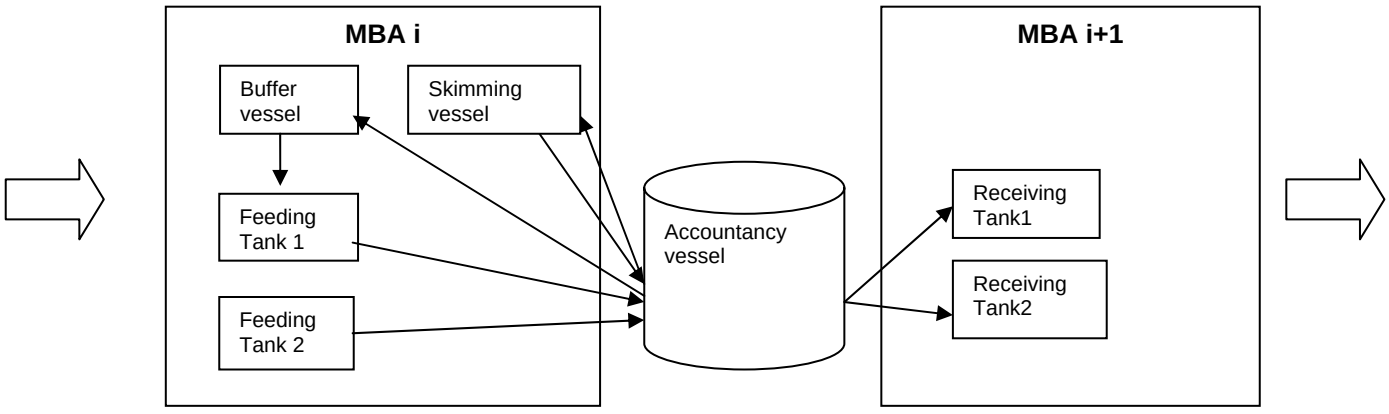



**Figure 1**: Exemple of Accountancy vessel environment.

### 2.2. Diagnostic Capabilities

Solution monitoring software provides system diagnostic. The software computes fissile material masses and propagates them across the process, but also identifies when and where the original event causing a material balance discrepancy is.

Two main approaches can be used for diagnosing disccrepancies : either a model of the plant is able to generate thes explanation by backtracking balance errors until a cause is isolated (i.e. the concerned transfer), or a model propagates mass computations along the actual process until a mismatch with operator allowed operations is found. The first approach is driven by a global detected discrepancy, the second is driven by raw data until a discrepancy is found.

The choice is made (backward or forward) depending on available tools, available data and a global philosophy in approaching this problem.

There has been significant work over decades to improve fault detection and diagnosis in large complex systems. Sometimes a full analytical approach is possible, sometime full heuristics are the only possibility. These two approaches can be combined into a so called hybrid approaches.

Model-based approaches are used on statistical process control, and formal method such as analytical redundancy have been researched deeply [3],[4].

The solution monitoring approach presented here is a heuristic and forward method adapted to transfer mismatch identification based on a symbolic representation of authorised operations. This requires configuration data for all the possible transfers in a sequence of tanks, transferred volume and duration of operations. The correct identification of a “cycle” - a complete and closed operation of a vessel - allows trigger computation of various indicator (boolean or values). Identification of discrepancies on accountancy vessel is therefore very important.

This method was designed in 1995 for Euratom Treaty implementation in AREVA NC La Hague. The mismatch detection method is generic and it has been used for monitoring Pu Canister production in La Hague, and has been adapted for the SMSS system at RRP for the IAEA Inspections.

### 2.3. Data abstraction level

The last facet of the solution monitoring problem for material balance comparison is the data organization. How is it possible to elaborate high level information about possible discrepancies while staying outside error boundaries of noisy signals in such huge amount of data generated by a reprocessing facility? The solution monitoring system needs then to run across abstraction levels.

| **Abstraction level** | **Relevant IT** | **Important Qualities** |
|---|---|---|
| 1- Measurement | Transceiver, Field buses | Robustness, Redundancy, Acquisition device calibration,<br>Share with Operator. |
| 2- Data acquisition | Industrial information architecture and network | Redundancy, Unit,<br>Share with the operator |
| 3- Equation and formulae | High constraints Archiving | Dedicated archiving, Embbeded formulas, |
| 4- Data interpretation | Expertise capture | Knowledge of Reprocessing facility<br>Near real-time |
| 5- Decision | Data structure archiving | Confidence,<br>Explanation |

**Table 1** : From data to knowledge

Table 1 above displays the abstraction levels of data: from 4-20mA analog or digital signal to the knowledge of nuclear material currently contained in an accountancy vessel with liquid transfer correlation through transfer circuits. Finally the comparison with Operator Declaration (OPD) can be automated.

For the solution monitoring in a reprocessing plant we can represent these level as shown in the figure here after (fig2.)

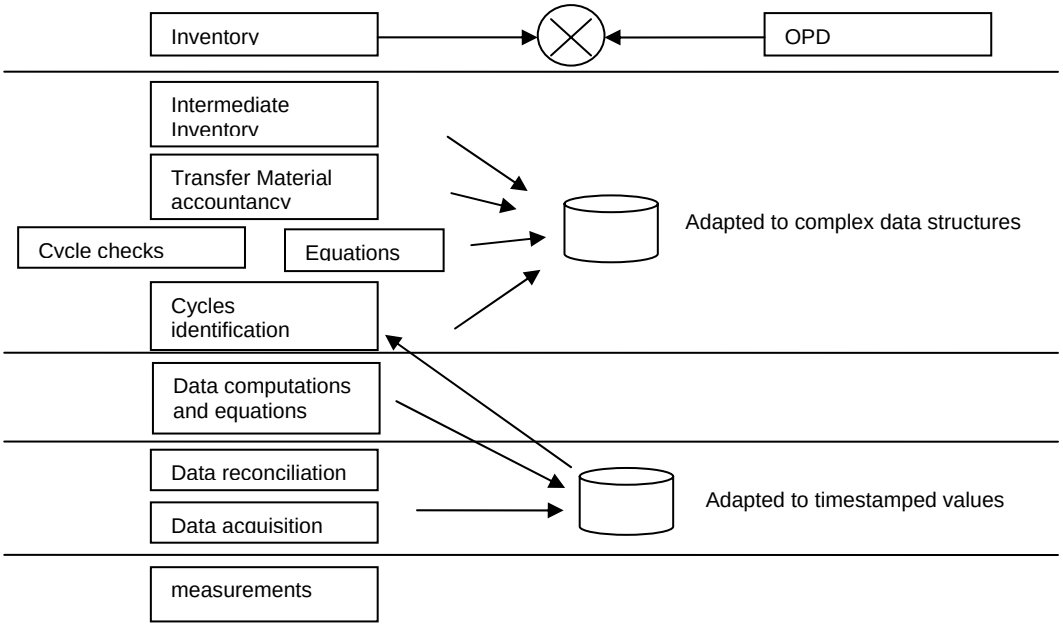



**Figure 2**: Solution monitoring abstraction levels

### 2.3.1. Measurements

Solution monitoring systems usually use a channel of operators sensors when available. Inspectors may have their own sensors too. In this case, these sensors must not interfere with operator activity : this can be source of discussions and complaints. Information redundancy can also be a reason for having inspectorate acquisition devices. The various cases are illustrated in Figure 3 and described below.

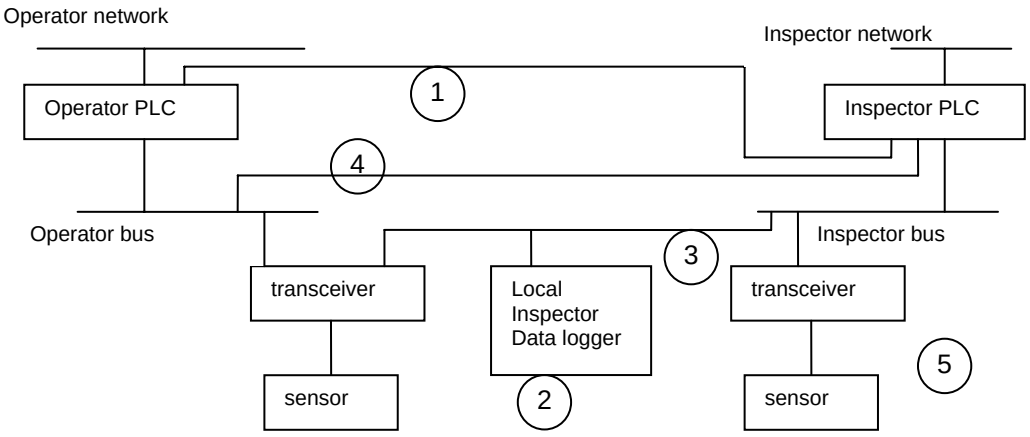



**Figure 3**: Different situations of measurement acquisition for monitoring system.

In Figure 3 we can see the main situations encountered: a measurement can be copied from the Operator Programmable Logic Controler (PLC) to the Inspector PLC (1), another transceiver channel may be used to send the data either on a local device (2) or to the Inspector bus (3), the data using the Operator bus can be copied to the inspector PLC (4) and finally Inspector can have their own sensors and industrial infrastructure (5).

### 2.3.2. Data acquisition

One important question concerns the industrial network or field bus. Inspection may require specific network or specific security solution when the network is shared (Fig 3 cases 1 and 4). If it is shared with the Operator then security must be enhanced to avoid a man-in-the-middle threat.

During data acquisition many events may generate interruption in the main data flow at Inspector network level (i.e. above measurement). To comply with this situation, redundancy or data reconciliation techniques can be useful. This is mandatory to avoid “holes” in the data which could correspond to interesting information. To avoid this, data acquisition procedures must rely upon redundant equipment, specific reconciliation procedure or robust software architecture

### 2.3.3. Data computation and equations

Data computation and formulae are applied on time stamped data. The result is also time stamped but to compute balance equations, the date when all necessary data are available for this may be “far” from acquisition date of the first variable. Therefore data archiving (raw and formulae results) is critical and it must manage times tamped data properly. To simplify data time coherence we can define an “analysis horizon” which is the time of the last complete and coherent set of raw data available to compute all formulae.

Most often, used formulae have their own valid ranges. It is mandatory to check these ranges before computing and generate irrelevant alarms in case of trouble.

One important challenge is then to store data in a database with a good time management. The system must archive a continous flow of industrial data while providing data to the analysis and interpretation level. In facilities having a lot of information in the balance area this is a real challenge, because, fast acquisition sampling provides the best computation accuracy rearding times tamped data.

### 2.3.4. Data interpretation

Data interpretation must facilitate diagnosis tasks for Inspectors. For instance, if an accountancy cycle is not coherent with the declared process then interpretation software must flag the event and indicate when it happens. This means that the data interpretation relies upon design information, especially allowed piping arrangement and vessel calibration that have been checked by Safeguard Inspectors [5].

In our approach the data interpretation is based on the identification of operating cycle of production equipment (by detecting start and end of cycle), cycle characteristics (start of filling from vessel X, start of emptying to vessel Y) for liquid transfer monitoring, the detection of material movement for canister monitoring or traceability of material movement in a review system for non-destructive assay.

This analysis and interpretation of time stamped data requires:

- Confidence in the time stamping procedure,
- Knowledge of possible allowed change in equipment operation,
- Knowledge of the facilities and the equipment operating cycles submitted to inspections.

*The last point is relevant to safeguard expertise and is critical to perform optimized an accurate safeguarding i.e. not spending time in analyzing specific behaviour that can be explained by operating constraints*.

## 2.4. Turning Diagnosis into Pattern-matching problem

In a forward data-driven approach for diagnosis, it is essential to detect any abnormal exploitation cycle that could invalidate balance equations. If abnormal conditins can be detected, in near real-time, it is not necessary to compute the balance equation.

The accountancy vessel cycle is the cornerstone of the approach. Vessels cycles are related because of the piping arrangement. Since only few of these piping arrangement are allowed it is posible to

describe all cases (low complexity). However it is mandatory to detect the piping arrangement in the data since the monitoring system is not aware of the selected arrangement for a given transfer. The detection makes use of raw data. The basic questions are: if a level change in a vessel what is the feeding or the receiving tanks? how is the density affected ? what is the temperature to be used in equations?

The goal is therefore to identify any abnormal exploitation cycles on monitored vessel without generating false alarms. Cycle identification is similar to a pattern matching problem on raw data related to level, density or temperature parameters. Once all linked cycles are identified then the "analysis horizon" can progress to run coherence verification to prepare balance equations.

The interest into pattern-matching techniques is driven by the fact that vessels are used repeatedly. Therefore we have pseudo-periodic signals with some particular interesting features such as an emptying transfer which can provide a reference point (for resynchronization) when the matching is lost.

However, cycle are not identical even for a given vessel. Transfers can be delayed or interrupted so it is difficult to look for a filtering (statistical approach) technique for pattern-matching. For the same reason template matching is not a good candidate for cycle matching. Among all candidate pattern-matching method [6] the syntactical method is the best suited.

Indeed, at the early beginning of this work (1995), we discovered that Euratom Inspectors used to focus on particular point of the cycle (in temperature, density or level). These points correspond to local behavior of the considered curve. The date of these occurences are used not only to perform computations whose results are used along the cycle but also to check the transfer flow of nuclear material.

Therefore these local cycle behaviors are of great importance for monitoring since they trigger computations and give the global view of a cycle including transfer correlation between vessels.

### 2.4.1. Syntactical Pattern Matching

Syntactical pattern matching considers the local behavior as pattern primitives or "words" and the link between them as the "sentence" describing the pattern. To illustrate this, consider the cycle level on an accountancy tank shown in Figure 4 (time along x axis) :

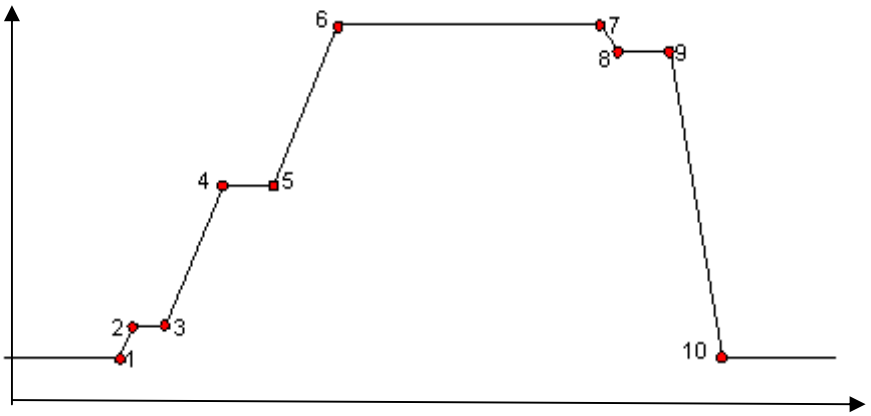



**Figure 4**: example of allowed cycle on acountancy vessel at UP3 in La Hague (level curve).

This pattern can be represented by the following sequence of local behaviors : start of skimming vessel return (1), then, end of transfer from skimming vessel (2), start of transfer from first feeding vessel (3), end of transfer from first feeding vessel (4), start of transfer from second feeding vessel (5), end of transfer from second feeding vessel (6), start of skimming (7), end of skimming (8), start of transfer to receiving tank (9), end of transfer to the receiving tank(10). No mixing is represented on this figure.

Considering that generally:

- with a transfer from only one feeding tank or,
- with a mixing after or between feeding transfers,

we still have a valid accountancy cycle, we can represent the category of allowed accountancy cycle by a graph linking the local bahaviour of the level curve.

We get then a unique description of all the allowed solution flows (a well-formed sentence). Here below is an example of such a sentence for UP3 accountacy vessel in La Hague.

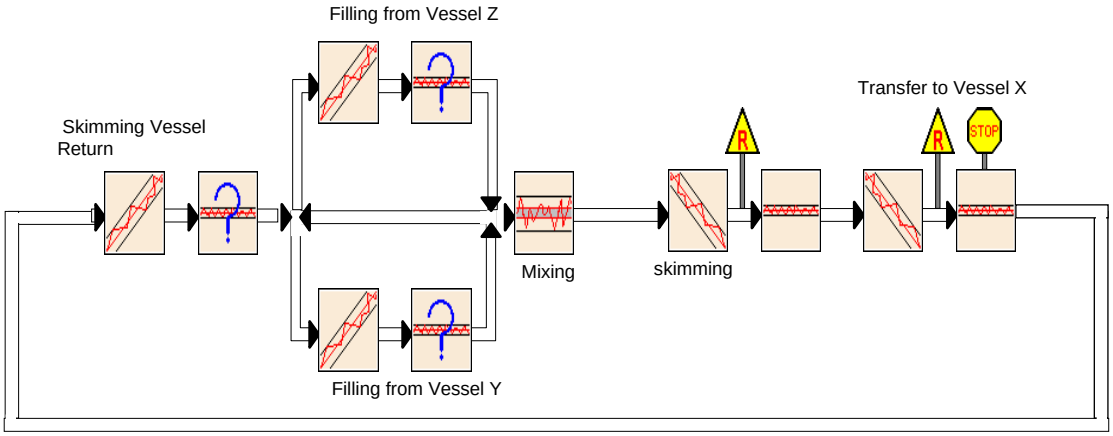



**Figure 5**: Syntactical description of the level profil of allowed accountancy cycles at UP3

In the example shown in Figure 5, the graph uses the level data. The definition of local behavior is so generic that we can scan level data and compute equations on density or temperature data depending on the attributes of the local behavior. We can then introduce dependence between computation (validity area etc) on different variables. This is very interesting for hybrid devices.

Each box corresponds to a local behavior of the curve and the graph gives the pattern to observe.

Links between boxes are compliant with a defined grammar. This latter is a set of rules that specify the transition conditions. These conditions depending on the fact that a local behavior is detected or not i.e. the status of the considered local behavior.

A state machine drives this dynamics by defining rules for status change for a local behavior. Each local behavior can have the following state:

- Ready (R) if the previous local behavior is in the “in Progress” state
- Idle (I) if the local behavior is not in Progress and it is not one of the next expected local behavior in the syntactical graph
- Monitored (M) if the previous local behavior has left the “in Progress” state but the conditions required for the considered local behavior are not yet verified.
- in Progress (P) if the local behavior has been detected.

The state transitions are managed as follows:

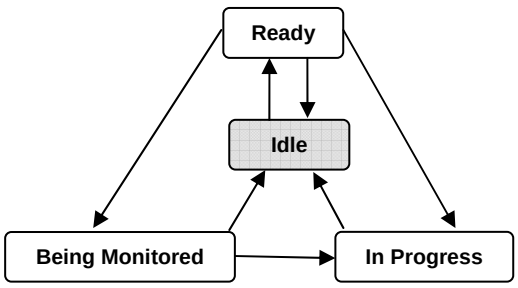


**Figure 6**: State transition graph for local behaviour

The normal state path for a local behavior is to start at (I) then to reach the (R) state and then (P). At that point if all criteria associated to this local behaviour are correct and the local behavior has been detected properly then it is declared *observed*. Nevertheless If one *time* criterion is exceeded for the next ready local behavior then the concerned local behaviour return to the idle state with an alarm (generally a timeout). Observation of a local behavior has of course a starting and ending time.

In a syntactical graph, a transition is a link between two states of two different local behaviors. Here one important thing to underline is the possibility that we have to link states between two local behaviors belonging to different syntactical graph each prototyping, the global behavior of two different vessels. This is the coherence check og the pipeline arrangement.

### 2.4.2. Running the pattern matching

The pattern matching behaves as follows[1]:

- an observation (or analysis) window is set along the considered signal. Its width is a parameter and the selected initial position is such that the first point $p_0$ of the signal is located on the left side of the window.
- The parameters searched are given by the current local behavior which is in (R) status. Let us imagine that this local behavior is a slope with a given tolerance.
- Signal points ($p_i$) inside the observation windows are examined: [$p_0$ $p_i$ ] slope are computed and checked with the target slope with the given tolerance.
    - If all segment [$p_0$ $p_i$ ] stay within criteria then the observation windows is moved to the next signal point and the local behavior stay in the P status ,
    - If there is signal point ($p_k$) for which the slope is out of the criteria, then the local behavior is no more observed and it reach the (I) status while the possible next local behaviors in the syntactical graph reach all the (M) status.

This loop is controlled by the so-called “analysis engine”.

The scanning of the signal with an observation window depends on the available signal points. These points are stored in the real-time database. This database storage has many implications:

- If every signal point is stored (even if the signal has no changes) then the observation windows will progress slowly with no real advantage for slope tests performance,
- If the recording process “over-samples” the acquired signal then we may discover slope change at a wrong date,
- A compromise has to be found regarding compression algorithm used for archiving. This compression must be such that no point is stored when signal doesn’t change, and it must store every point for interesting part o the signal.

Some COTS[2] products offer these capabilities in industrial application. The configuration of their deadband is also critical.

### 2.4.3. Triggering of Computations and Synthesis

Local behavior also has attributes: some are generic and others are specific. These attributs can be organized to manage set of constraints. Here below is a sample of the constraint list :

- Temporal constraints:
    - Minimum Duration: the minimum duration of a correct detection
    - Maximum Duration: the maximum duration of a correct detection
    - Exceeding Criteria Duration: the maximum duration for a local behaviour to be temporarely out of criteria while in (P)
    - Latency Duration: the mandatory waiting time for the transition M to P

[1] This is a simplified description
[2] Commercial On The Shelves

    - Waiting time before alarm: the maximum duration between previous local behavior observation end and the beginning of the observation of the next local behavior.
- Computation constraints:
    - Thresholds (high, medium, low) which correspond to slope parameters.
    - Target value
    - Set of possible value
    - Return for date: used when the system needs to go backward to find an accurate date for the detection.
- Specific constraints: related to local behavior type.

Computation can also be triggered by the detection of a behavior. For example, evaluation of DeltaV (which is the volume variation) between feeding and receiving vessels are automated with the detection of a slope local behavior. Finally every requested data for material balance can be computed once a cycle is detected and closed and once concentrations are all available.

## 3. Application to solution monitoring at AREVA NC La Hague plants

The integration and commissioning of the solution monitoring SYSTEM7 for UP2 and UP3 took place in 1996-1997. An important requirement was that the hardware and software architecture should be the same for both reprocessing plants. Nevertheless, these plants are not identical. For instance the accountancy vessels are not located at the same places in the plants. Moreover, for network reasons, it was mandatory to have a local PLC at spent fuel unloading facility (T0) of UP3 plant.

It was then decided to copy the UP3 SYSTEM7 hardware architecture to UP2 despite the latter could have been less complex. Nevertheless, this situation is better for maintenance purposes of SYSTEM7 at UP2 and UP3, and also, for an easier spare parts management of both plants.

Rearding the accountacy vessel location, it has been decided that the SYSTEM7 should be as much flexible as possible through parametrization.

### 3.1. SYSTEM7 Architecture

The intial hardware architecture was the following:

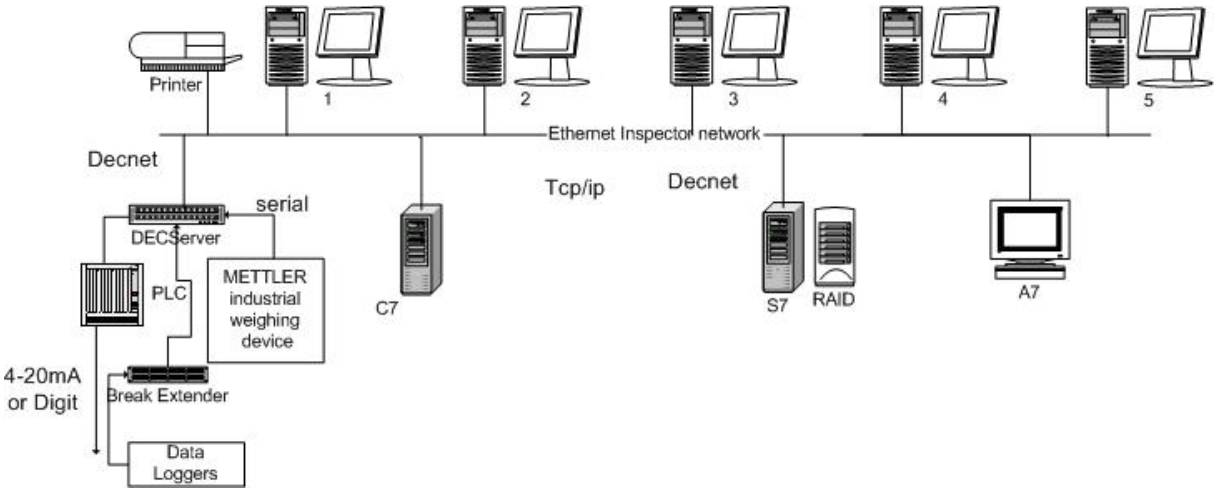



**Figure 7**: SYSTEM7 Hardware architecture as designed in 1995

The concentrator C7 for UP3 is located at T0 which is far from the remaining part of SYSTEM7 located at DG-TREN Inspector Office at UP3. The concentrator reads data from PLC and other equipments via a hub which is in the same cabinet. These data are copied on the server S7 which support the data base. A workstation for analysis (A7) is used to perform the signal analysis and to perform the necessary computations to prepare intermediate inventory.

Digital Corp Computers (DEC) hardware was selected in 1995. A DecServer provided the equipment interfaces. A housing for specific circuit was made because of the necessity to extend a 'break' signal for the dialog with the data loggers. The C7 was a DEC2000, the S7 was a DEC2100 and A7 was a DEC800.

The solution required a multi-layer software architecture. To deal with possible network problem on UP3 a local Oracle database was installed on C7 to allow autonomy during network outage or upper level crashes. To feed the database with data coming from the DecServer using DecNet protocol, we used the DEC Bstar product. A batch job running on S7, periodically copied new data on S7. The Oracle database on S7 stored one year of data (from the last inventory) on line. The A7 workstation ran a G2 expert system to handle alarms from discrepancies in the cycle identification using syntactical pattern matching or related to computation errors. Inspectors had 2 months of on line raw data and analysis results with Inspectors comments and corrections.

The Y2K situation brought a problem because Bstar was not maintained anymore. Moreover G2 version for Alpha processor on DEC hardware was also abandonned by the vendor (Gensym Corp). After a migration to Windows NT for Alpha, Gensym decided to not support this G2 version. This necessitated an upgrade of the system hardaware and software. The system was migrated to Windows NT and Intel CPU.

The database was changed from Oracle to PI (from OsiSoft). This brought better raw data performance since PI's compression algorithm was much more efficient that the one we implemented ourselves on C7. PI brought easier production routines since it is more robust than the previous architecture especially with the possibility to have remote interface as near as possible to the equipment.
Moreover, the flat-file organisation of the data storage allow to manipulate backup as simple as drag-and-drop operation. Network interruption and S7 crashes did not lead to any lost data. With these database there are no online limitations for data scanning and database "crawling".
The consequence is that today we can continue to merge peripherical equipment on the SYSTEM7, making this architecture the main Inspector on site centralized system while extending the amount of available online data (see Figure 8 here after).

As a result, Bstar, DecNet, Proc C developments (compression and data transfers), Oracle Backup layers disappeared in the simpler and more flexible software architecture. However an ACCESS or SQL Server data base was included to store high level data structures.

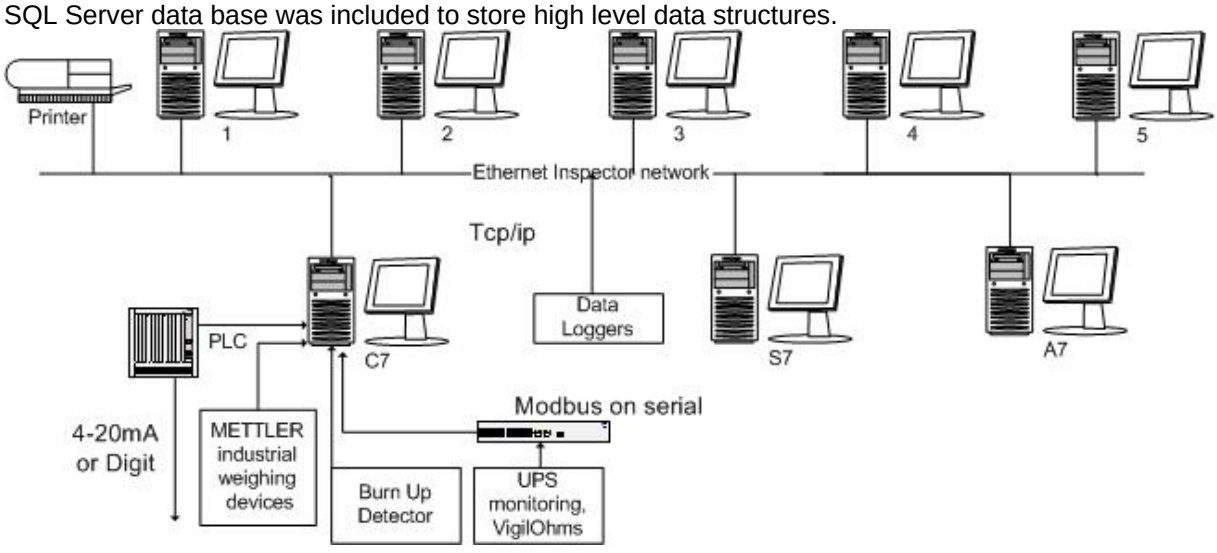


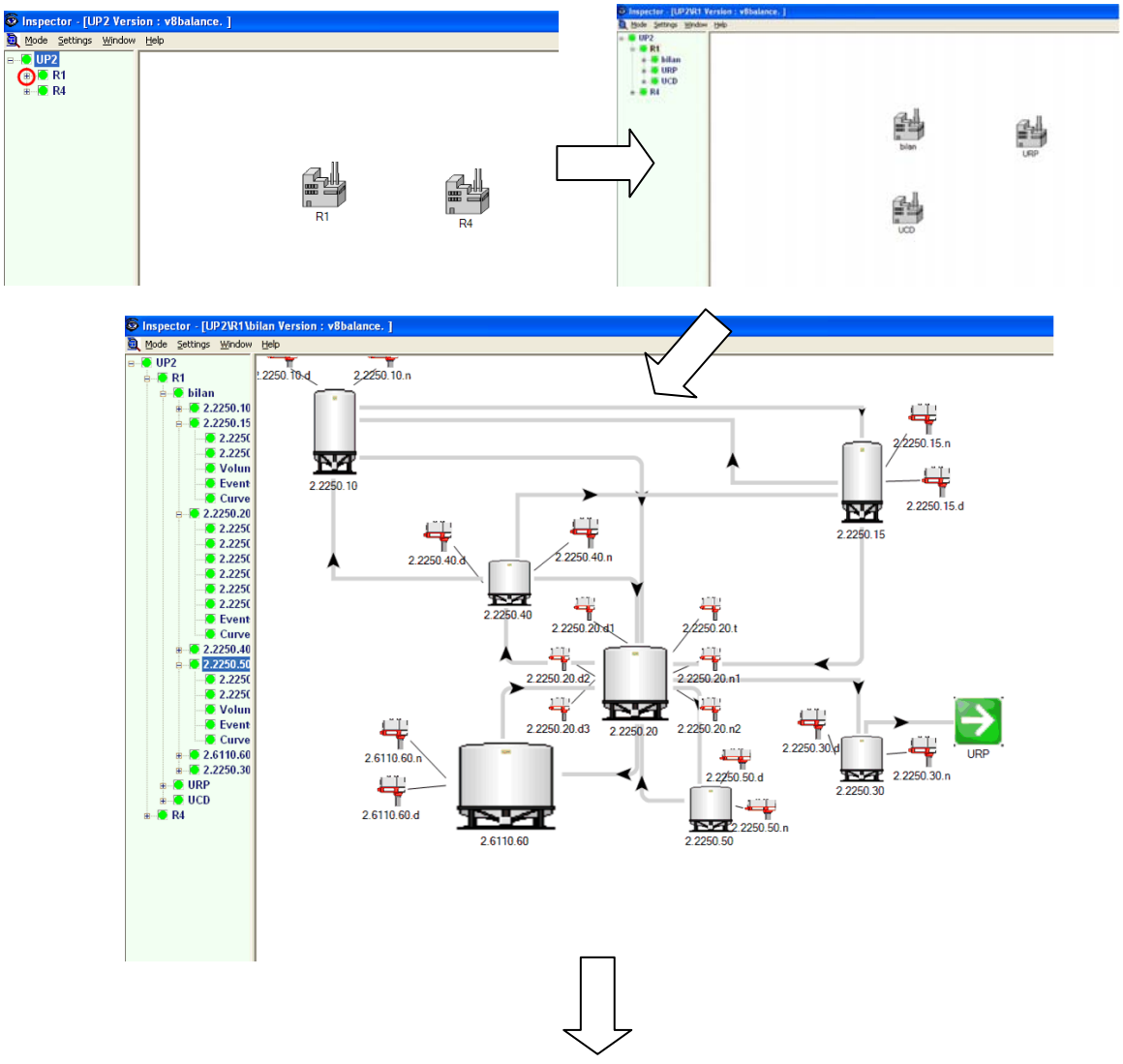

**Figure 8**: The current SYSTEM7 architecture with added devices

Finally due to high level cost for the G2 licenses maintenance (3 licenses was needed for UP2, UP3 and Maintenance Plateform SYSTEM7s), DG-TREN entrusted Joint Research Center (JRC) in ISPRA to develop its own analysis engine based on the experience gained with SYSTEM7 A7. This led to a software product named Data Analysis and Interpretation (DAI) which continues to evolve especially regarding connection to other data historian and exporting capabilities.

### 3.2. Data Analysis and Interpretation

The DAI software is under the responsability of JRC ISPRA at the NUclear SAFEguard Unit (NUSAFE). This software have been developed by Euriware with requirement of JRC and DG-TREN and run the syntactical pattern-recognition as described above ([7], [8]). We give here below some example of screen shots.

In Figure 9 a and b we see some capabilities of the system (here the drill down capability). First, synoptic view of the selected part of a plant and finally the selection of a sensor of interest or the selection of all signals from a vessel (density, temperature, level). The tree structure on left side of the screen allows fast navigation among equipment of interest. All signal data are in mA.



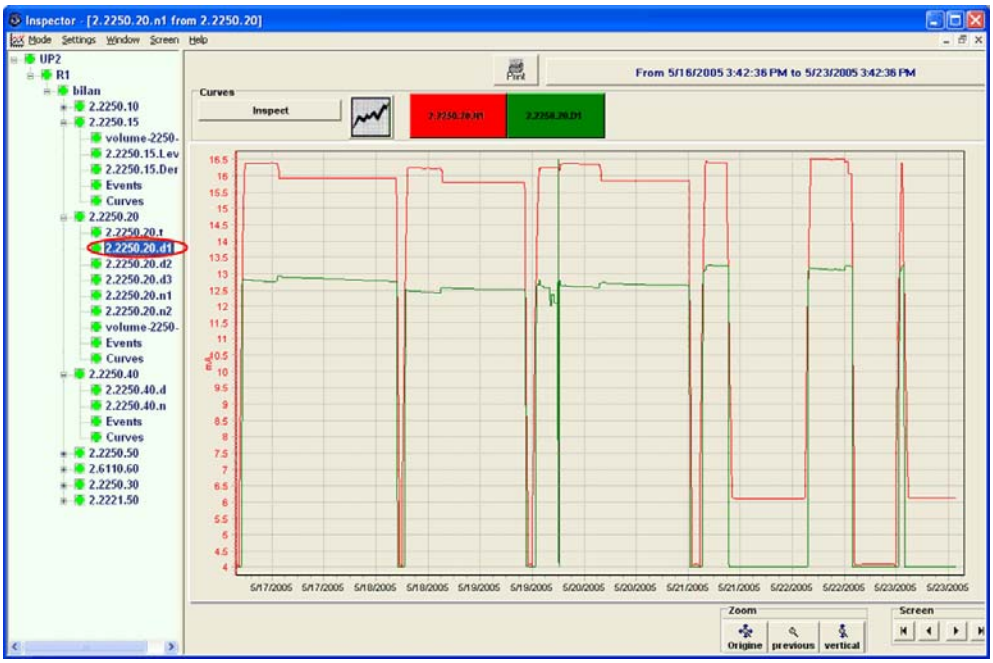



**Figure 9 a):** Drill down to look at level and density of accountancy vessel.

We also have the capability to assign sensors to vessels or tank by a simple dragg&drop action from the left-hand side to the main screen.

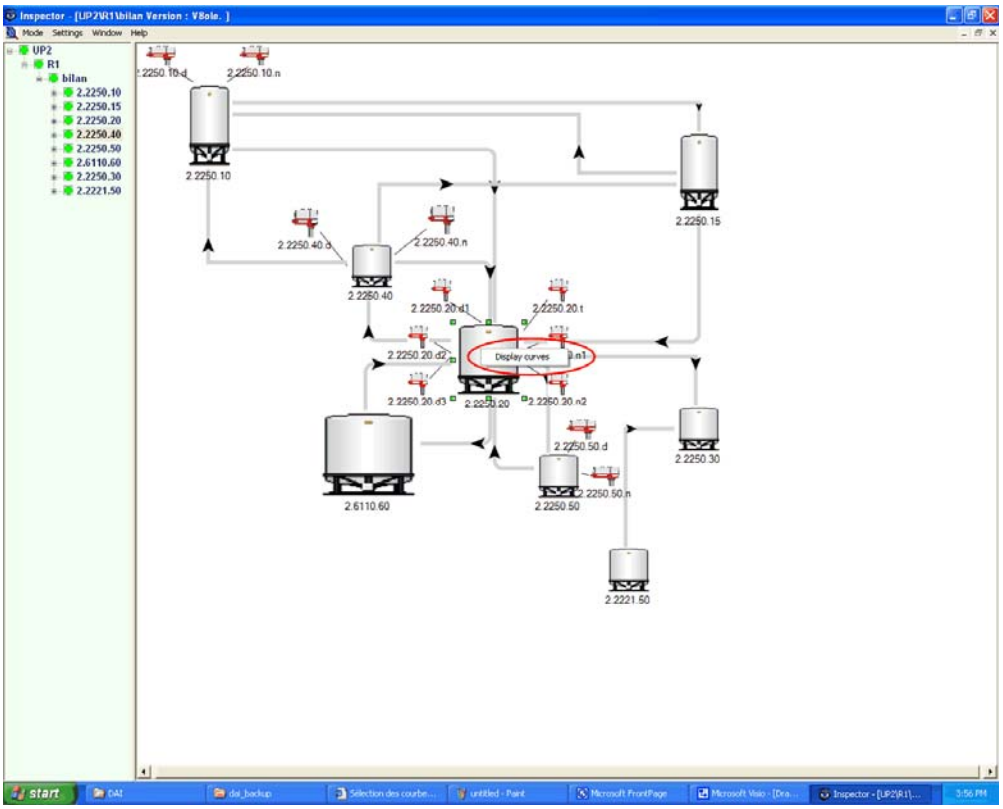



**Figure 9 b)**: Getting all data for a given vessel on one click.

Figure 10, shows events detected on level curve which triggered an alarm. The Inspector has the possibility to modifiy the event attributes and to launch the analysis again. Most of the alarms are related to a wrong date for the event leading to incoherence in the following computation process.

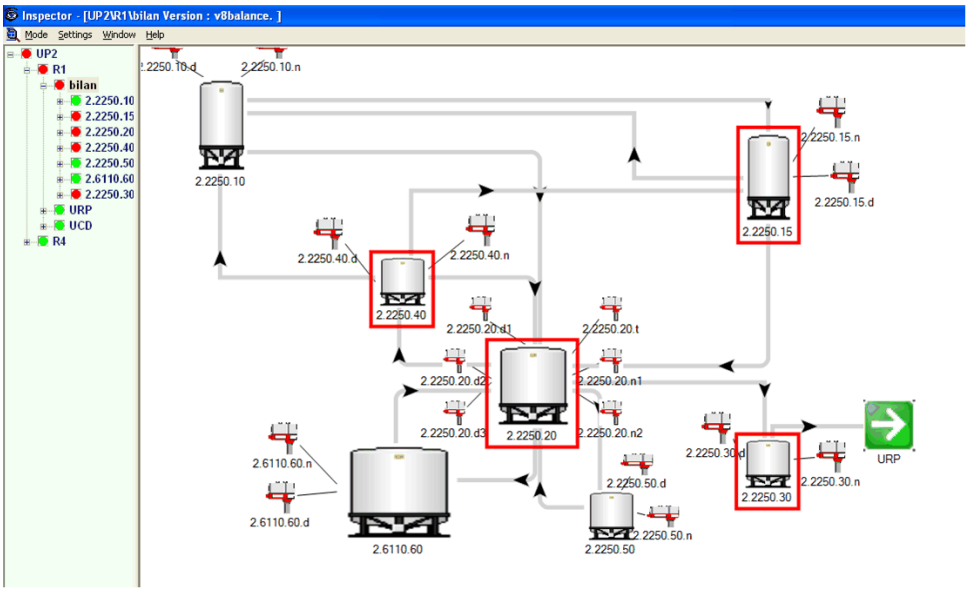


**Figure 10**: Finding an alarm with alarm localisation on vessel synopsis

A deliverable of DAI is the value table associated to a cycle. Figure 11 shows a value table which displays an alarm because of flow coherence problem.

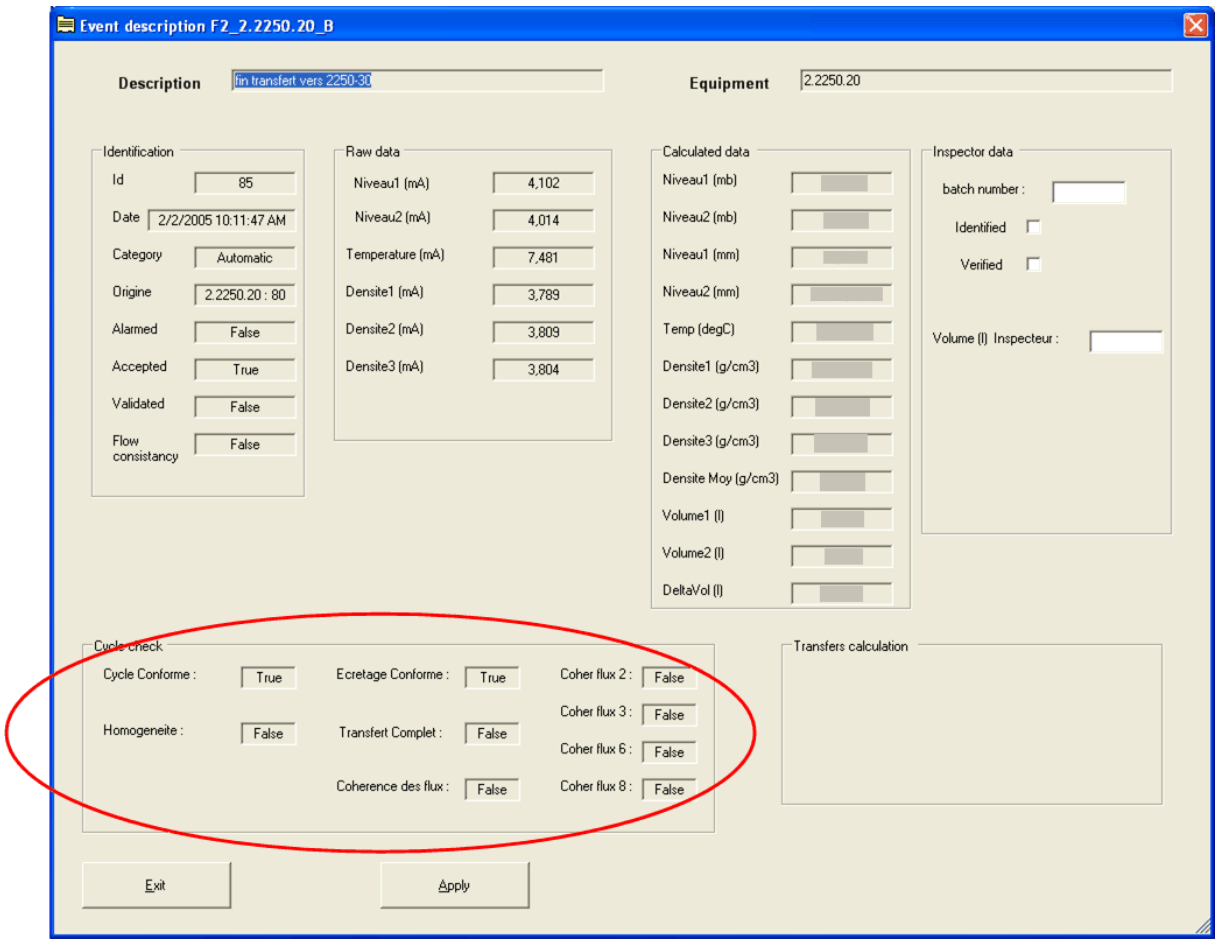



**Figure 11**: The alarm on event, once selected, shows the contextual data and can apply Inspector modification to be propagated in the computations

### 3.3. Extension to other device monitoring

DAI is also used to monitor an industrial balance, in the Pu canister production glove box. The signal is captured on the printer port of the Mettler balance (5 of those are monitored on UP2, 2 at UP3) and is brought to the C7 through a PI interface listening the serial link. Since this signal is periodic we used the DAI to monitor the operator activity and to measure the Pu mass output from the facility.

The SYSTEM7 now includes the Uninterruptible Power Supply (UPS) signals and also the signal from the burn-up detector (BUD).

### 3.4. Parameterization

The DAI software needs a parametrization procedure allowing easy parameter testing. It is posible to run analysis on raw data in a “playback” mode to provide the Inspector with a validation functionality.

The parameterization consists of defining the cycle pattern, then in assigning values to different attributes. Due to the great number of attributes (especially those related to time management) it can be useful to have help and to assist the Inspector. The DAI software provides help in the configuration procedure by predefining values in some specific cases. This is done from a view of the signal and from a defined syntactical graph: the user defines area of interest on the plot view and assigns area to local behavior on the graph. The system then triggers computation for parameterization.

The advantages for having several attributes to manage time for event detection, is that the SYSTEM7 can be adapted to new authorized configurations. Modifying a few attributes can integrate the new figure in the analysis engine.

Figure 12 here below shows the DAI parameterization environment.

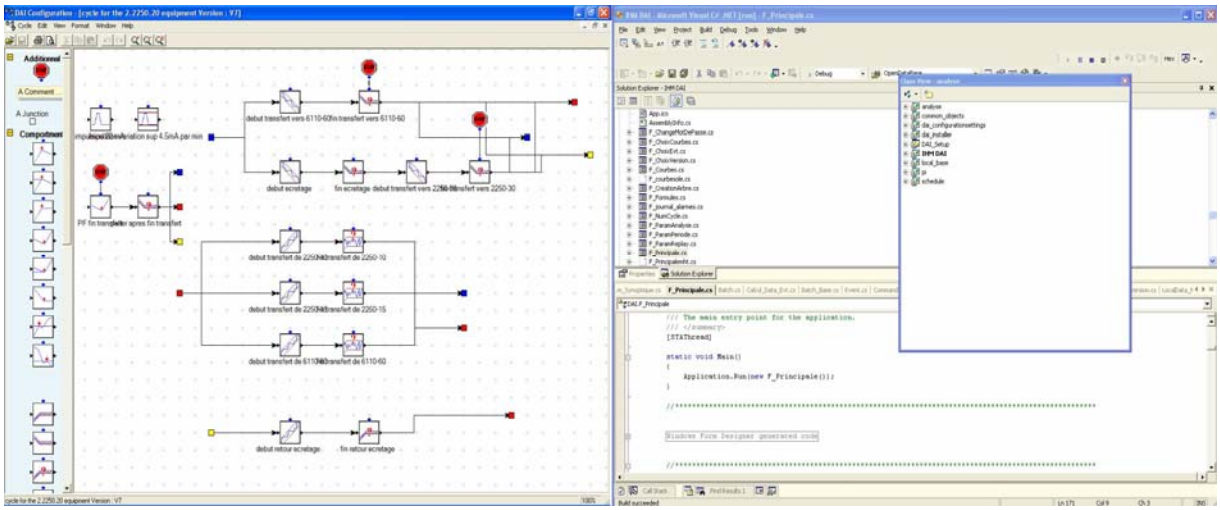


**Figure 12:** Parameterization environment for DAI

## 4. Application to Solution Monitoring at JNFL RRP

The IAEA developed slightly different software for the solution monitoring at RRP (the Solution Monitoring Software System SMSS). The differences include:

- It is a web-based application,
- the GUI is based on a different Inspection approach,
- equations are hard coded (they are parameters in DAI but at RRP they can be complex algorithms, and the estimate of concentration at the extraction cycle could not be made using declarative formula),
- some behaviors are specific (the mixing for instance),
- the Inspector can see all transfers in a table,

- the hard-coded algorithm allows Pu concentration estimates in unmonitored equipment such as the Pu extraction cycle,
- Pu concentration is estimated from density and acidity (user parameter) this allows concentration propagation along the flow,
- SMSS can work only on volume or mass-volume signal (DAI engine is working on raw 4-20mA).
- the recovery point located at the end of a DAI cycle, is replaced by a series of two detections
- the SMSS monitors 92 vessels to be compared to the 13 vessels for UP2 (40 other tanks are used to prepare intermediate inventory but they are not monitored).

The SMSS relies upon a hardware and software architecture which is more complex since:

- it encompasses all applications for Inspections,
- the IAEA relational database is the cornerstone of the IAEA infrastructure.
- it shares data with JNFL which require the two network to be connected
- IAEA specific cabinet send raw data using the JNFL network which require a security procedure to ensure data integrity.

### 4.1. Architecture

The most important difference is the RRP data evaluation architecture. The SMSS is a subsystem of the Integrated Inspector Information System(I3S) and relies on the I3S file system and relational database. The I3S infrastructure supports many other Safeguards application and SMSS was designed to use this infrastructure.

The I3S infrastructure architecture is shown in figure 13. In this schematics we can see tha ta connection is made with the JNFL Network, and with the server for the on-site laboratory. This is quite different from the current La Hague monitoring since only raw measurement data are shared on SYSTEM7.

A web-server is also needed since all Inspector applications user interfaces are web-based on this technology. These two characteristics require the use of firewalls to ensure protection of the De-Militarized Zone (DMZ).

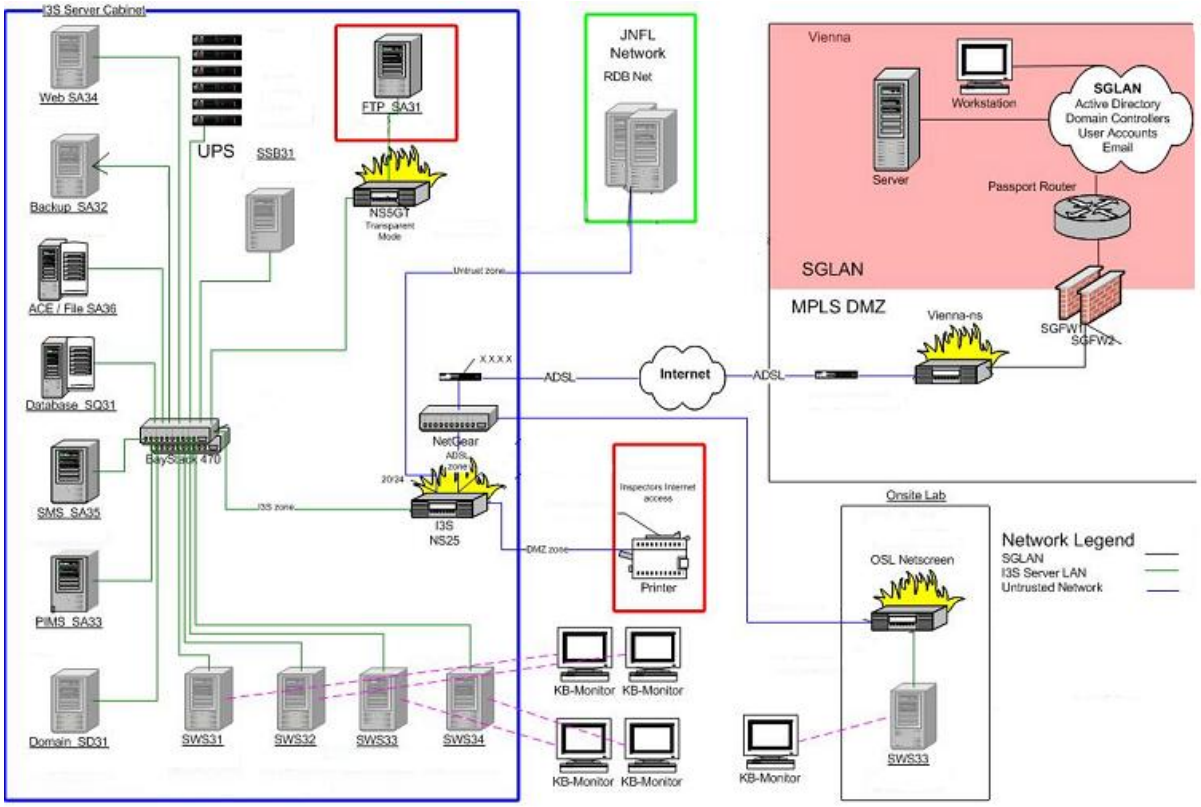


**Figure 13:** I3S Architecture

This architecture distributes computation among multiple servers. It has the advantage of separate applications with weak coupling. For instance, a dedicated application is used for alarm management and equipment state of health. The applications send alarms with standardized calls to an alarm service. The application coupling is done through a central relational database. With a good tuning of the database, the activity of one application has little impact on others. With a well-dimensioned network the database request trafficking is not critical.

## 4.2. SMSS Analysis and Interpretation

The IAEA database is the cornerstone of the software architecture. However, data contained in ascii file are available on another server. These ascii files come from IAEA cabinet for measurement acquisition. In normal mode data contained in files are copied into the database. However in some situations the ascii files can not be parsed and copied into the database (for instance for a database shutdown or a network problem). In such a case, when the system recover, there are “holes” in the data and the files must be interpreted to get all data. SMSS is in charge of preprocessing those files, which is also a difference with the DG-TREN SYSTEM7.

Figure 14 below shows the top level window for solution transfer evaluations. This screen complies with IAEA standards for web-based application that enables easy browsing between I3S applications. This screen is a synthesis of the plant solution status. Events are color coded to allow inspectors to focus on possible alarms (in red). The event color coding is the result of an “evaluation procedure” which is similar to the analysis engine of DAI.

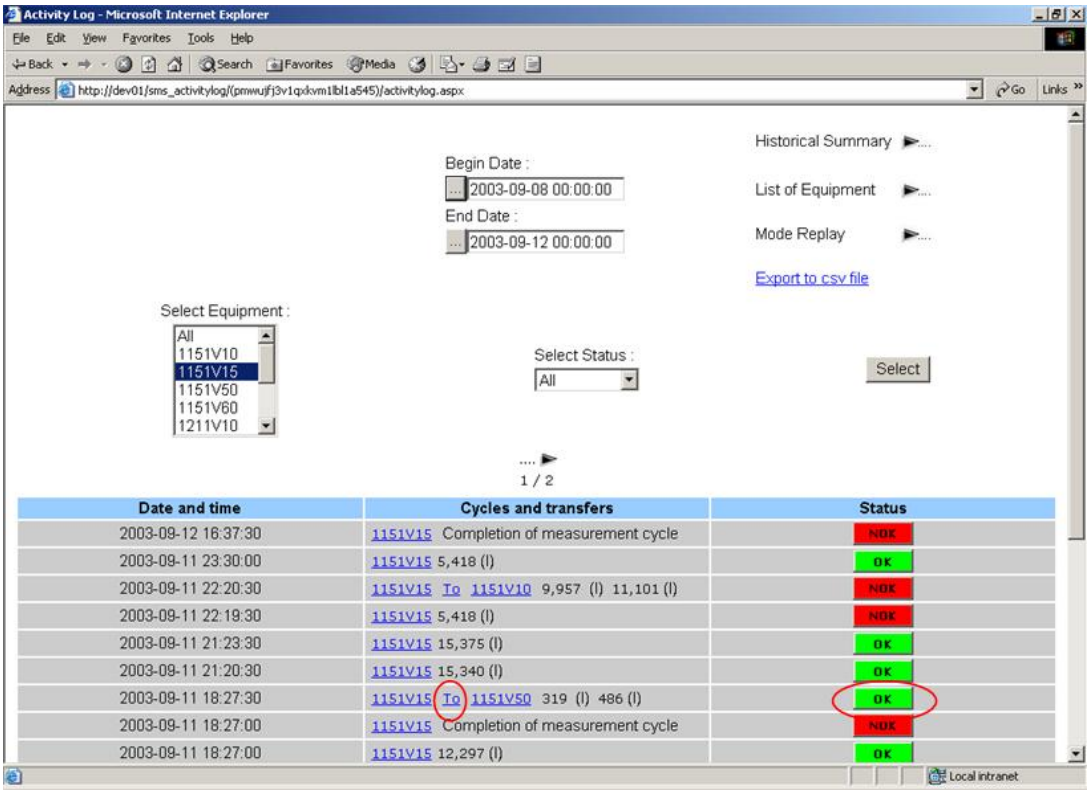



**Figure 14:** SMSS Top Level GUI Window.

In this example, equipment numbers are hyperlinked. The user can click on a vessel to drill down to the raw data (as for the DAI software). The Inspector can then have the detail as shown Figure 15.

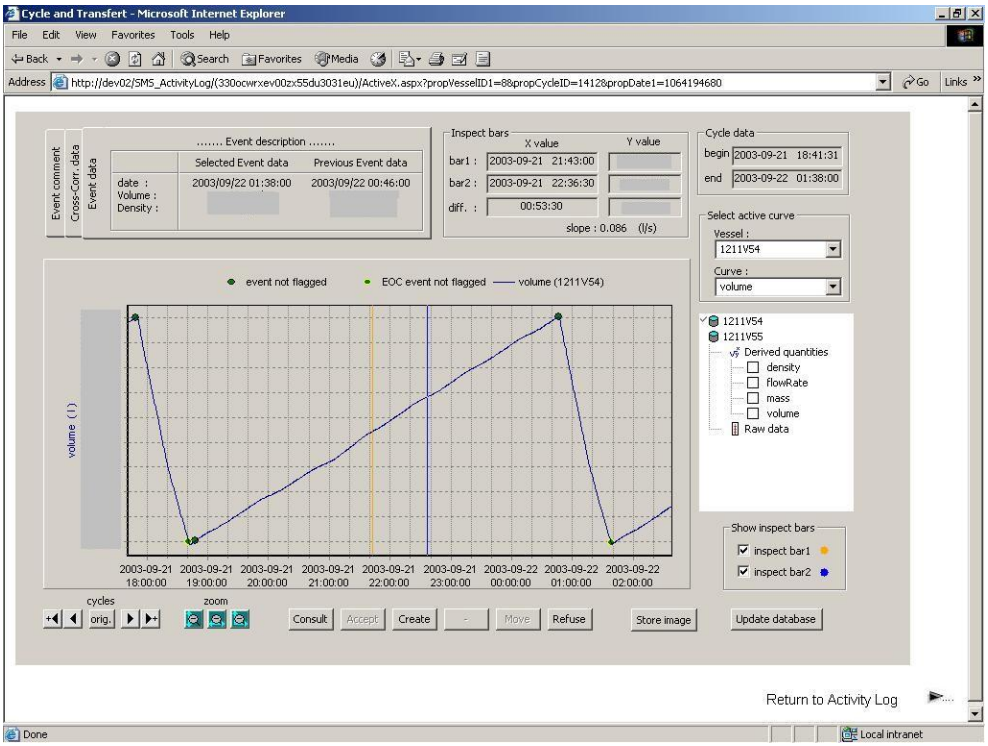
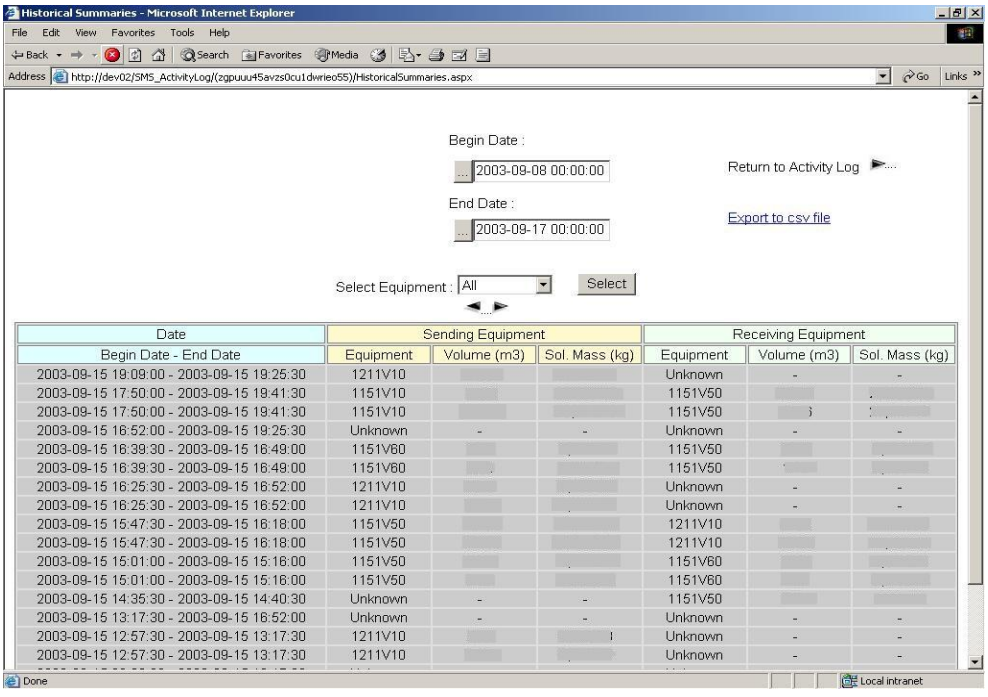



**Figure 15:** SMSS Event Display Window

The Inspector has the capability to see every transfer between two dates as shown in Figure 16.




| Date | Sending Equipment | | | Receiving Equipment | | |
|---|---|---|---|---|---|---|
| Begin Date - End Date | Equipment | Volume (m3) | Sol. Mass (kg) | Equipment | Volume (m3) | Sol. Mass (kg) |
| 2003-09-15 19:09:00 - 2003-09-15 19:25:30 | 1211V10 | | | Unknown | - | - |
| 2003-09-15 17:50:00 - 2003-09-15 19:41:30 | 1151V10 | | | 1151V50 | | |
| 2003-09-15 17:50:00 - 2003-09-15 19:41:30 | 1151V10 | | | 1151V50 | | |
| 2003-09-15 16:52:00 - 2003-09-15 19:25:30 | Unknown | - | - | Unknown | - | - |
| 2003-09-15 16:39:30 - 2003-09-15 16:49:00 | 1151V60 | | | 1151V50 | | |
| 2003-09-15 16:39:30 - 2003-09-15 16:49:00 | 1151V60 | | | 1151V50 | | |
| 2003-09-15 16:25:30 - 2003-09-15 16:52:00 | 1211V10 | | | Unknown | - | - |
| 2003-09-15 16:25:30 - 2003-09-15 16:52:00 | 1211V10 | | | Unknown | - | - |
| 2003-09-15 15:47:30 - 2003-09-15 16:18:00 | 1151V50 | | | 1211V10 | | |
| 2003-09-15 15:47:30 - 2003-09-15 16:18:00 | 1151V50 | | | 1211V10 | | |
| 2003-09-15 15:01:00 - 2003-09-15 15:16:00 | 1151V50 | | | 1151V60 | | |
| 2003-09-15 15:01:00 - 2003-09-15 15:16:00 | 1151V50 | | | 1151V60 | | |
| 2003-09-15 14:35:30 - 2003-09-15 14:40:30 | Unknown | - | - | 1151V50 | | |
| 2003-09-15 13:17:30 - 2003-09-15 16:52:00 | Unknown | - | - | Unknown | - | - |
| 2003-09-15 12:57:30 - 2003-09-15 13:17:30 | 1211V10 | | | Unknown | - | - |
| 2003-09-15 12:57:30 - 2003-09-15 13:17:30 | 1211V10 | | | Unknown | - | - |

**Figure 16:** Display of all transfers in a selected period

## 4.3. Parameterization

SMSS provides a specific stand-alone application for parameterization like DAI. Inspectors may want to change parameters while the system is performing an evaluation. To test new configurations, the Parameterization tool provides a validation function based on the new data. As for DAI, a playback mode is available.

The SMSS reference signature pattern is declared and parameterized in a user-friendly application. The validation performed with real data (with read only access) and an animation of the reference signature. This is shown on Figure 17 and 18 below.

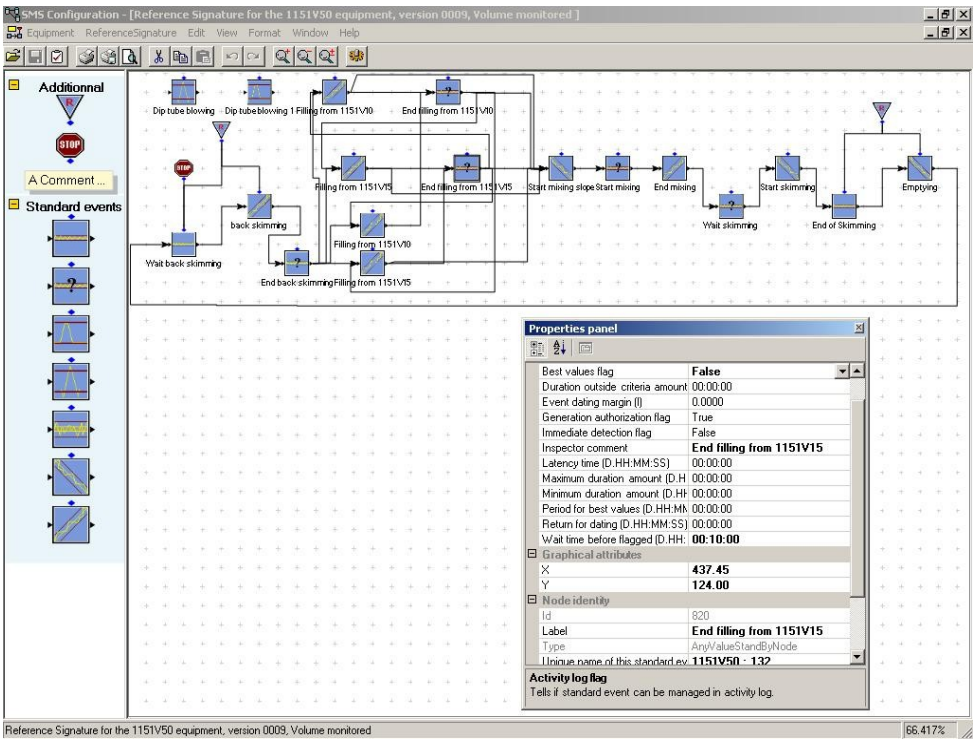



**Figure 17:** Parameterization of a cycle

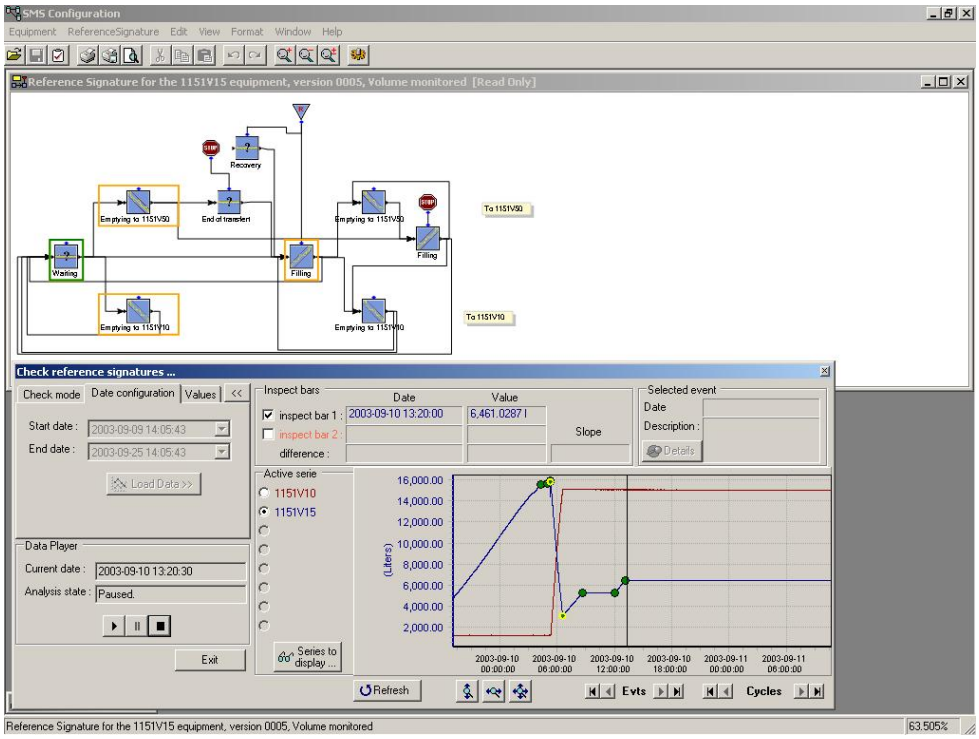



**Figure 18**: Running parameter set validation

## 5. Bubbling System for volume measurement

Solution monitoring will not work without good measurement of the solution volume. To make these measurements, process knowledge and the capability to a design simple, robust and accurate method is mandatory.

This is illustrated with a short description of bubbling system as the one Euriware is currently developing for the MAYAK RT1 Reprocessing plant in the framework of a TACIS project led by DG AIDCO and DG JRC.
A bubbling system uses three probes. In each probe air flow at a rate of about 7Nl/h and the pressure is monitored. The end of one probe is located over the solution. This probe measures the reference pressure within the tank (tanks containing nuclear material are generally under a negative pressure to avoid any contamination.
For accountancy vessels we have the following situation (see fig 19). Based on the three measured pressure Pr (reference), Pl (level) and Pd (density) and the temperature T

- the density $\rho$ is given by $\rho = K'(Pl - Pd)$ ) with $1/K' = g\,i$ where g is the gravitational constant and i the distance between the tips of the level and density probes
- the level "h" with respect to the bubbling probe (not the tank bottom) is given by $h = K''(Pl - Pr)$

The volume is a 4th order polynomial of h, generally obtained after calibration. This formula can be computed on the PLC or at upper level.

The most important aspects for this type of system are:

- what is the quality of the industrial air ?
- what are the requirements in terms of an eventual decontamination of the equipment ?
- how to avoid any contamination which might result of a tank pressure higher than the air pressure used in probes?
- What are the requirements for availability of the data?
- What are the requirements in terms of accuracy and precision of the measurements?

The systems being delivered monitor 11 vessels. They use WIZCON as Supervisory Control And Data Acquisition (SCADA) for surveillance as shown in Figure 20. Stainless steel cabinets are shown in Figure 21, have been designed and built to comply with decontamination requirements.

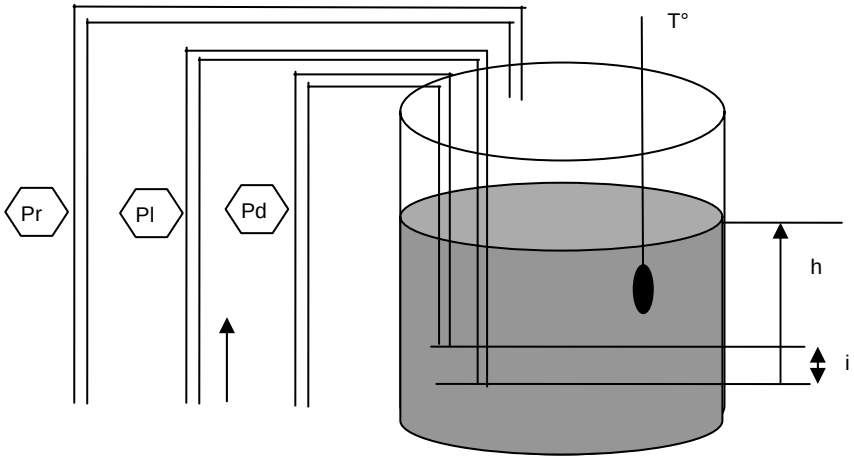



**Figure 19**: Bubbling system

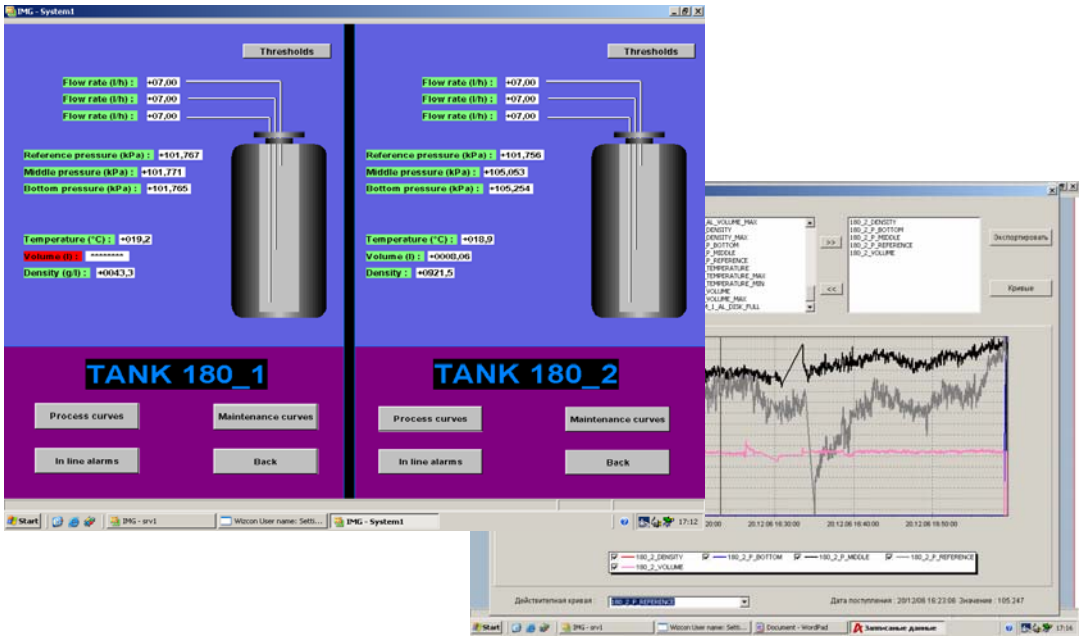



**Figure 20:** Wizcon SCADA system

**Figure 21:** One of the bubbling systems for MAYAK Reprocessing plant

## 6. Non destructive assay review (NDAR) at RRP

Since most of pseudo-periodic signal can be modeled using syntactical patterns, it is possible to imagine applications that monitor these kind of signal. This has been successfully tested on DG-TREN monitoring of $PuO_2$ Canister production in La Hague and for the Burn up detector in La Hague.

The IAEA decided to use this approach to monitor non-destructive assay data at RRP. Here the idea is to automate the evaluation of cyclic and standardized operation of devices placed along the reprocessing lines and used to trace the fissile material movement. For instance a situation where a set of fissile material goes one way along the reprocessing line and then goes back must be automatically detected and reported. When cycles are validated then computations are performed as with the solution monitoring applications.

The challenge here is the various kind of equipment monitored. Each measurement system has its own behavior. These systems include:

- Integrated Spent Fuel Verification System (ISVS)
- Integrated Head-end Verification System, Feeding and Shearing Component (IHVS-FS)
- Integrated Head-end Verification System, Leached Hulls Component (IHVS-LH)
- Rokkasho Hulls Monitor System (RHMS)
- Temporary Canister Verification System (TCVS)
- improved Plutonium Canister Assay System (iPCAS)
- Directional Canister Passage Detectors. (DCPD)
- MOX Storage Containment and Surveillance (MSCS)
- Vitrified Canister Assay System (VCAS) with normal, measurement and hatch abuse cycles
- Waste Crate Assay System-A (WCAA)

The Los Alamos National Laboratory (LANL) developed an event generator which works on threshold (and not on slope as with solution monitoring). The events are not sequenced. The NDAR system checks for LANL event ordering and then checks for compliance of measure cycles on iPCAS and RHMS. NDAR computes measurement windows which are passed to LANL-developed components for computation. Results are recorded in the I3S database.

The NDAR Graphical User Interface (GUI) shows a matching list (fig 22). This list indicates the status of the comparison, for a given assay, between NDAR computation and operator declaration (OPD). The operating cycle of the assay subsystem is classified as NORMAL or OTHER. The NDAR computation results are shown (Pu mass here). The table entries are hyperlinked to corresponding data from related subsystems. For instance, the related On-Site Laboratory (OSL) results can be displayed by clicking on the table entry. The left-most column displays the matching status between the OPD and the NDAR results. Clicking on an alarmed button drills down to the raw data that the match result is based on. Figure 23 shows a display of IHVS-LH raw data to allow inspector review of the underlying data.

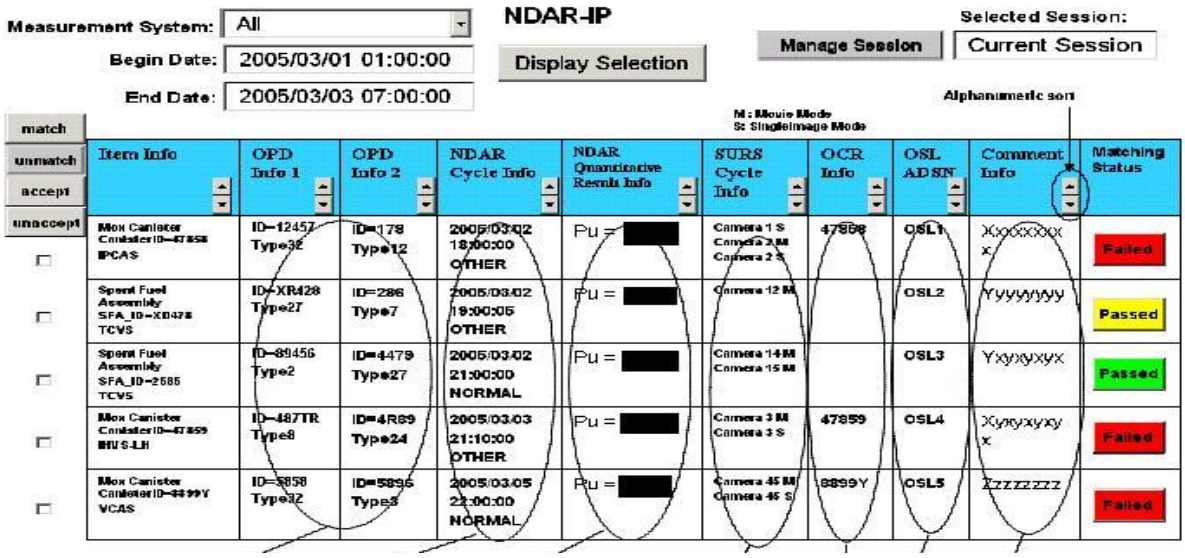



**Figure 22:** NDAR matching list interface

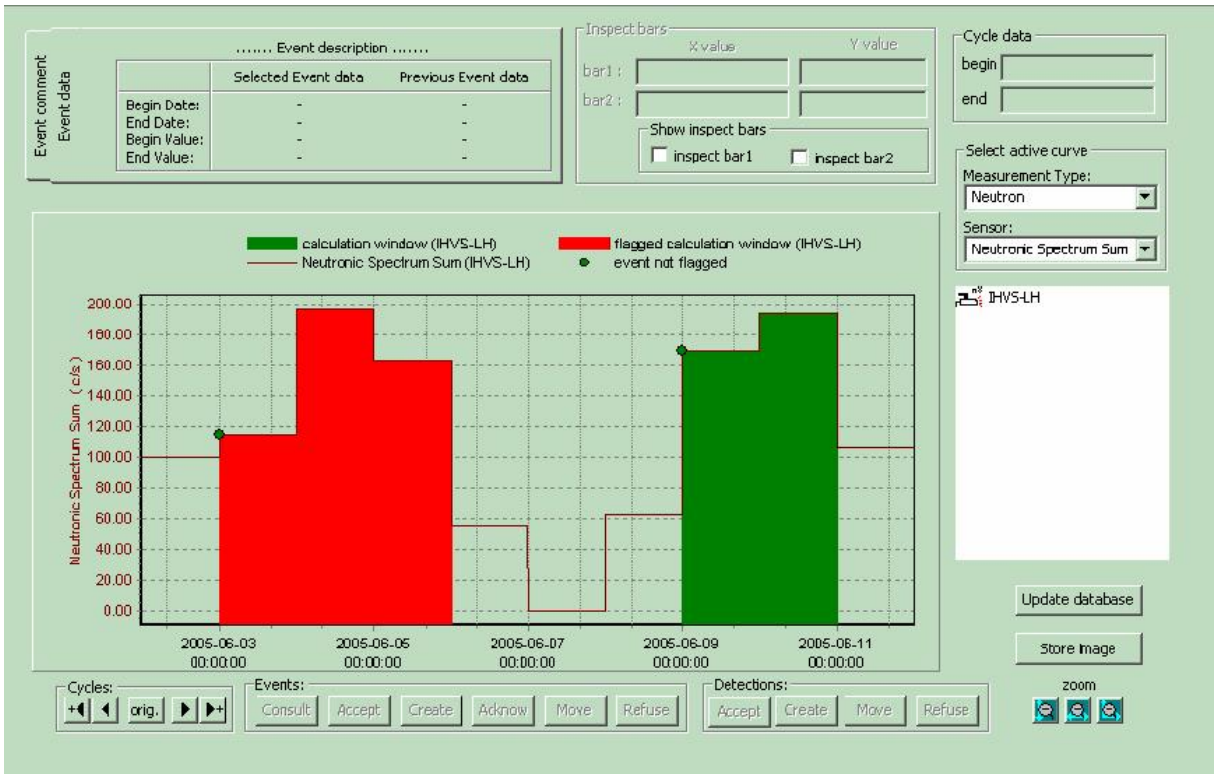



**Figure 23:** Example of display for a neutron counting device

The NDAR data evaluations are parameterized using the same approach as the solution monitoring systems discussed above. This gives the parameterization window shown in Figure 24 below.

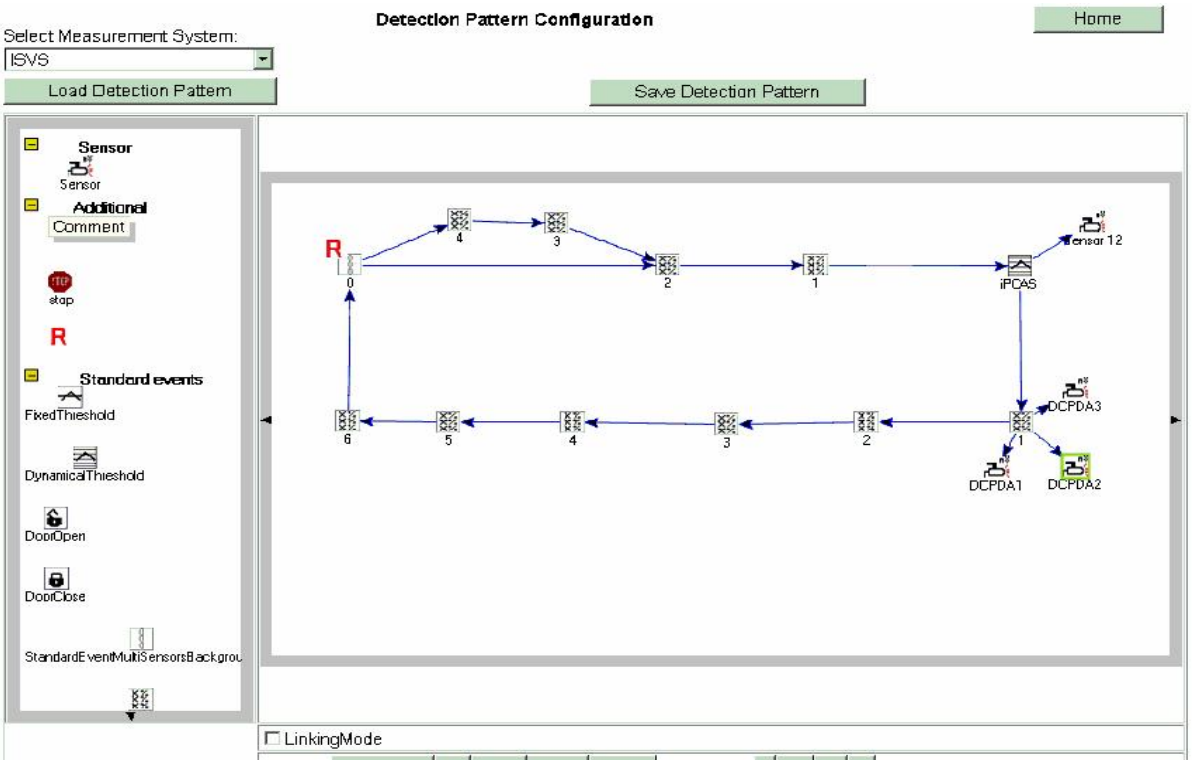



**Figure 23:** Parameterization of ISVS device.

## 7. Conclusion

Solution monitoring is the cornerstone of safeguarding wet reprocessing facilities. This paper presented two approaches for solution monitoring and shows how these tools can be extended to evaluate NDA data, or other cyclic operating equipment from a reprocessing facility.

Solution monitoring systems assist Inspectors by providing a tool that ensures the coherence of flow (small deltaV) according to authorized transfers, the computation of U/Pu mass from monitored equipment based on volume, temperature and density and any kind of comparison among periodic or cyclic signals. These results can feed applications for interim inventory and provide estimates of the material balance before the inventory period for safeguards inspectors.

The data-driven diagnostics provide all the necessary contextual information to allow inspectors to understand the plant processes and correct discrepancies. For instance a transfer can generate an alarm and the system leads the inspector to the local behavior causing the alarm. The systems can detect deltaV errors due to a bad positioning of events in time.

This paper describes two approaches to solution monitoring:

- The original system which initially relied on a “small” architecture using COTS components such as expert system (G2) and relational database. This system became more robust by simplifying the architecture and using a data historian database (PI) in addition to a relational database for event storage. The current trend is to integrate other applications together with SYSTEM7. The DAI system is a very light Windows application that can also be setup for small system (it has been installed at TOKAI on a small architecture). In this approach, PI performs the data acquisition. Using another historian like Matrikon may need a hardware study want to ensure robustness.
- The newer system, SMSS, provides a web-based user interface and uses a central relational database for data from SMSS and other safeguards applications. This approach brings a maximum data coherence among the data and the integrated safeguards applications. Data acquisition problems with raw solution monitoring data are dealt with by specific pre-processing procedures.

Both approaches have shown their capabilities in dealing with other problems than solution monitoring. For DG-TREN SYSTEM7, the Burn Up Detector is now analyzed by DAI, and for the IAEA this technique has been used for NDA data processing and evaluation. .

Solution monitoring software can be delivered with specific measurement devices and can be connected to low level measurement systems. Most often this equipment already exists and requires a shared network. Numerous security technologies ensure the integrity of the network and the measurement data transmitted on it.

The application of SMSS and NDAR suggest that several strategic axes could merge. Near-real time accountancy, non-destructive analysis and solution monitoring bring a complete near-real time view of plant operations and material accountancy that focus on both flow management and material movement but also perform MBA input/output calculations to provide near real-time on line inventory for the plant.

## 8. Ackowledgment

This work is the result of many years of experience with these technologies or concepts. These experiences were captured during industrial project with hard constraints.

We would like to acknowledge:

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- Mr. Olivier Lengronne; Euriware; who is in charge of the SYSTEM7 maintenance in UP2 and UP3,
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- Mrs. Shirley Johnson (who left IAEA) who was responsible of the RRP project at IAEA/SGOA,
- Mr. Mike Ehinger (who left IAEA ) who was SMSS project leader in RRP team of SGOA in the Agency,
- Mr. Florin Abazi (who left IAEA) who installed and started up on-site the SMSS)
- Mr. Willem Janssens (Head of NUSAFE unit, IPSC, JRC ISPRA).

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# IAEA Verification of Military Research and Development

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***Abstract:***

*Any state intending to manufacture nuclear weapons must do more than acquire a sufficient quantity of fissile material. It must also undertake an extensive research and development programme to design and manufacture a usable device—a process known as weaponization. Current International Atomic Energy Agency safeguards focus on the accountancy and control of fissile material. Much less emphasis is placed on detecting clandestine weaponization activities. This paper analyzes the Agency's prospects for uncovering a clandestine weaponization programme, if it were given the legal authority to do so.*

*First, the Agency's legal authority to verify research and development is discussed. It is argued that the Agency does have the right to inspect suspected weaponization activities where nuclear material is involved. If no nuclear material is present, its mandate is much less clear. In either case, however, the Agency lacks the authority to collect evidence on which it could base the request for an inspection.*

*A strategy for detecting clandestine weaponization activities is then discussed. A simple model for the weaponization process is presented and various indicators and signatures associated with different stages in the weaponization process are outlined. Relevant verification techniques are also discussed. The detection of hydrodynamic testing is used as an illustration. It is shown how the IAEA might be able to detect such tests through various means including environmental monitoring, satellite imagery and the analysis of export data.*

**Keywords:** weaponization; safeguards concepts

## 1. Introduction

Although acquiring fissile material is the main challenge facing a state intending to manufacture a nuclear weapon, the difficulties associated with weaponization should not be underestimated [1]. As Iraq discovered prior to 1991, designing, manufacturing and testing the various components needed for an implosion weapon is a complicated task [2]. Iraq began a determined weaponization programme in 1988. Three years later, when its efforts were uncovered following the first Gulf War, it had failed to complete the development of either an initiator or explosive lenses. The Iraqi case is not unique. The process of weaponization has invariably proven more time consuming and complicated than anticipated.

This observation raises an interesting possibility. Could the International Atomic Energy Agency (IAEA) be given responsibility for detecting clandestine weaponization efforts as a means of helping to prevent proliferation [3]? Currently, IAEA safeguards focus on the fuel cycle. The IAEA is tasked with verifying the non-diversion of declared nuclear material as well as the absence of undeclared nuclear material. Such safeguards will always be the primary barrier to proliferation. It is inevitably easier to detect a diversion from a declared facility or even the existence of a clandestine facility than the small-scale research and development activities associated with weaponization. Nevertheless, the verification of military research and development may have a useful supplementary role to play. Most

importantly, if a potential proliferator is able to bypass the need to develop enrichment or reprocessing technology by buying or stealing fissile material, then detecting weaponization efforts may be the only way to prevent proliferation. Even if the evidence is not strong enough to prove that a state is developing nuclear weapons, it might still act as a stimulus for the IAEA to increase its scrutiny of a state's fuel cycle activities—past and present.

There would, of course, be a number of disadvantages to the IAEA attempting to detect weaponization activities on a routine basis. It would be an expensive, intrusive and difficult process. Unambiguous evidence of weaponization would be hard to come by, complicating the process of drawing conclusions. Ultimately, however, this debate would be moot if weaponization activities were too unobtrusive for the IAEA to be able to detect them. This paper attempts to analyze the feasibility of detecting a clandestine weaponization programme. In particular, it outlines a strategy for doing so and assesses the chances of success. First, however, it briefly addresses the question of whether the IAEA's legal mandate entitles it to attempt this task.

## 2. Does the IAEA have the legal right to investigate weaponization?

The Agency's legal right to investigate suspected weaponization activities has been discussed in depth elsewhere [4]. Only a brief summary is presented here. The most fundamental question is whether weaponization activities prior to the assembly of a nuclear weapon are prohibited by the 1968 Nuclear Non-Proliferation Treaty (NPT). Although the treaty prohibits the 'manufacture…[of] nuclear weapons or other nuclear explosive devices' by non-nuclear weapon states, it does not define this injunction any further. During negotiations, the 'Foster Criteria' were put forward to provide a definition of the word 'manufacture' in the form of general purpose criteria [5]. They state that 'facts indicating that the purpose of a particular activity was the acquisition of a nuclear explosive device would tend to show non-compliance. (Thus the construction of an experimental or prototype nuclear explosive device would be covered by the term "manufacture" as would the production of components which could only have relevance to a nuclear explosive device.)' The Foster Criteria have not been challenged so far, and support the view that the term 'manufacture' refers to more than just the final assembly of a nuclear weapon [6].

Should the Agency obtain evidence of weaponization activities *involving nuclear material*, it could investigate them by means of a special inspection. More controversial are the Agency's rights in regard to weaponization activities not involving nuclear material. Former IAEA Director General Hans Blix has argued [7] that in such cases the Agency would have the right to request a special inspection as the activity would provide evidence that the state was intending to divert nuclear material. The current Director General, Mohamed ElBaradei, has been more cautious in arguing [8] that 'the Agency's legal authority to investigate possible parallel weaponization activities is limited, absent some nexus linking the activity to nuclear material.' This issue is related to the question of whether the IAEA is responsible for verifying all the injunctions in article III of the NPT or just a subset of them.

Although the Agency does have the right to investigate at least some types of weaponization activity, it would be challenging—within its current mandate—to obtain evidence on which to base the request for an inspection. There is a useful analogy to be made with 'traditional' safeguards on nuclear material. Before the advent of the additional protocol, the Agency had the legal right to inspect a facility at which it was suspected that clandestine nuclear activities were taking place—but it did not have the tools necessary to detect such activity. Indeed, even today in states without additional protocols in force, the Agency does not claim to be able to verify the absence of undeclared nuclear material. Similarly, with the tools currently at its disposal it is unlikely that the Agency could obtain evidence of clandestine weaponization activities—making the question of whether it could inspect them moot.

## 3. A verification process for weaponization activities

There is at least one key difference between searching for clandestine weaponization activities and searching for clandestine fuel cycle activities. Information provided by states is the starting point for verifying the absence of undeclared nuclear material. For example, inconsistencies within a state's declaration could indicate the existence of undeclared nuclear activities. In contrast, a state that does have a clandestine weaponization programme is very unlikely to reveal incriminating information in its declaration.

Searching for a clandestine weaponization programme therefore involves attempting to detect signatures and indicators of particular weaponization activities (the same principle that is currently employed to look for clandestine fuel cycle activities). The distinction between a signature and an indicator is a conceptually useful one. Indicators are the inputs to a process—specialised items of equipment, raw materials and design information, for example. Signatures are the outputs from a process, such as effluent, radiant energy (e.g. in the form of the appearance of a facility) and finished products.

### 3.1. A simple model for weaponization

A first step in mapping out the signatures and indicators associated with weaponization is the development of a model, i.e. a list of the processes associated with manufacturing a nuclear weapon. Ignoring the overarching political process, the main steps in the manufacture of a simple, first generation fission device are as follows [9]:

1. First, a state must develop a **design strategy** on the basis of its preliminary understanding of nuclear weapon physics. The development of a nuclear weapon is an iterative process and the design strategy will almost certainly be refined in the light of experience.
2. For an **implosion device** a state must
    a. develop and characterize suitable high explosive (HE);
    b. design, test and manufacture HE lenses.
3. For a **gun-type device** a state must
    a. design and characterize an appropriate propellant;
    b. design, test and manufacture a gun-type assembly.
4. To **manufacture the metallic components of a warhead** having first acquired suitable fissile material in metallic form, a state must
    a. characterize the neutronic and other physical properties of the material;
    b. acquire and test the technology required to fabricate the pit;
    c. fabricate the pit.
5. To **manufacture the non-nuclear components** of a warhead (for example the initiator, tamper or reflector, firing sets and safing, arming, fuzing and firing devices) a state must
    a. develop and characterize appropriate materials (e.g. polonium-210 for an (α,n)-initiator);
    b. acquire or develop other relevant items of equipment (e.g. neutron generation tubes);
    c. design, test and manufacture the components.
6. Finally, a state must **assemble the warhead**.

There are a few important caveats about the model. First, it excludes the steps that would be required to obtain fissile material in metallic form since those are already covered by IAEA safeguards. Second, because it is based on information in the public domain this model is necessarily simplified and incomplete. Clearly a practical strategy to detect clandestine weaponization activities must be based on a much more sophisticated model. Third, it may be possible for a proliferator to miss out some of the processes listed above if it receives help from a state or individual with knowledge of nuclear weapons design, or by using information that is already in the public domain (for instance the neutronic properties of uranium-235). In spite of these reservations the model is useful for illustrating the key ideas associated with the verification of military research and development.

| Activity | Equipment | Effluent | External physical features |
|---|---|---|---|
| Fabrication of fissile pit for an implosion device | Glove boxes or hot caves with special ventilation systems; remote manipulators; remote loaded environmentally-controlled furnaces (plasma, e-beam, induction or electric); NC multi-axis turning, grinding, milling or combination machines; NC non-wire electrodischarge machines; NC coordinate measuring machines (linear, angular or combination) | WGPu, plutonium oxide, tantalum, magnesium oxide, aluminium, graphite, calcium fluoride (for a plutonium pit)<br><br>HEU, uranium oxide, graphite, zirconium silicate, magnesium silicate (for a uranium pit) | Physical security |
| HE development | Hot isostatic presses; NC multi-axis milling machines; firing sets (e.g. exploding bridge wire detonators) | Suitable HE (e.g. baratol, cyclotol, RDX) | Physical security; expansion of an existing ordnance facility; lightning protection |
| Hydrodynamic testing | Pin domes; high speed oscilloscopes and recording devices; flash X-ray generators; flash X-ray recording systems (photographic, digital or analogue); framing or streak cameras; explosion containment vessels | Depleted uranium, natural uranium, lead, tantalum, other plutonium stimulants, tungsten, beryllium | Physical secuity; test pad; housing for equipment; sand bags; exclusion zone; physical effects (visual and auditory) of HE tests; control room (possibly underground) away from test site |
| Fissile material experiments (including initiator development) | High-speed neutron counters; neutron generator tubes; hot cells or glove boxes | HEU, WGPu, beryllium, plutonium-238, polonium-208, polonium-210, actinium-227, radium-226 | Physical security; heavily shielded/underground laboratory; laboratory physically isolated from control room |
| Reliability testing | Electrodynamic vibration test systems; vibration thrusters | | Physical security; tests of new designs of freefall bombs or missiles |
| Preparations for a full nuclear test | Drilling rigs; neutron, X-ray and gamma ray detectors, scattering stations and cameras; streak or framing cameras; high-speed oscilloscopes; coaxial and fibre-optic cables | | Physical security; mining operations; large-diameter pipes laid out; road construction; remote site |
| Weapons assembly and preparations for the storage and deployment of weapons | | | Especially high levels of security; new patterns of military activity (e.g. aircraft practising for nuclear weapon delivery) |

**Table 1:** Selected weaponization activities, their signatures and indicators. Note that the list of equipment given in the first row relates to the manufacture of a plutonium pit. Fewer items of less sophistication would be needed to manufacture a uranium pit. KEY: HE=high explosive; NC=numerically controlled; WGPu=weapons grade plutonium; HEU=high enriched uranium

### 3.2. Indicators and signatures of weaponization

Table 1 summarizes some of the main indicators and signatures associated with weaponization [10]. It includes a couple of activities beyond the scope of the model which could nonetheless be useful, namely preparations for a nuclear test (the means is already in place, of course, to detect an actual test) and preparations for the storage and deployment of nuclear weapons. The table is certainly not exhaustive. Not only are there other indicators and signatures associated with the activities listed in the table, but other weaponization activities might also be detectable. For instance, a political decision to start nuclear weapons development might result in the recruitment and training of scientists (quite possibly in non-military programmes) and consequently changes in the structure of the scientific community.

Almost all (if not all) of the indicators and signatures listed in table 1 are ambiguous [11]. That is, they are relevant to activities other than the manufacture of nuclear weapons. There are legitimate reasons, for instance, for wanting to acquire new types of high explosive. The discovery of a single ambiguous indicator would not prove that a state was intending to manufacture nuclear weapons. Nevertheless, strong evidence could arise from detecting a number of ambiguous indicators and signatures. Detecting clandestine weaponization activities on a routine basis would require an algorithm to assess the significance of detecting combinations of indicators and signatures, as well as the development of a data management system so that correlations and associations can be spotted.

### 3.3. Verification techniques

Signatures and indicators, such as those listed in table 1, are only useful if the means is available to detect them. Table 2 lists various verification techniques and the types of weaponization activities they might be useful in uncovering [13]. Although the Agency is currently empowered to use a number of the verification techniques listed in table 2, it is not entitled to use all of them. Because of the small-scale nature of research and development into nuclear weapons, convincing evidence is only likely to come from combining the results from a number of different data sources.

This form of verification emphasises the collection of information and the role of judgement in assessing it. It is a significant departure from the Agency's traditional safeguards approach (i.e. the approach it took before Programme 93+2). This approach, which was primarily designed to detect the diversion of nuclear material from declared facilities, was highly mechanistic. Searching for clandestine weaponization activities by detecting correlations of indicators and signatures does, however, bear some strong similarities with the Agency's information-driven approach that has accompanied the introduction of the state-level approach and integrated safeguards [12]. The Agency's experience of information driven safeguards would be invaluable in developing a system to search for clandestine weaponization activities.

Nonetheless, evidence of clandestine weaponization activities is likely to be weaker than evidence related to the misuse of nuclear material. The interpretation of such evidence could, from a political perspective, be a very sensitive exercise. In spite of article 19 of INFCIRC/153 which enables the Board of Governors to make a finding of non-compliance if 'upon examination of the relevant information…[it] finds that the Agency is not able to verify that there has been no diversion of nuclear material', the burden of proof for proving non-compliance has, in reality, always fallen upon the Agency. If the same demanding standard is to be applied to the detection of clandestine weaponization activities, it would be very hard for the Agency to present sufficient evidence to justify action.

The hardest part of detecting a weaponization programme is likely to be uncovering the preliminary evidence on which to base further investigation. Such preliminary evidence could come from a number of sources. Information provided by member states is likely to be useful because of the intelligence resources that states have at their disposal. Moreover, under article VIII.A of its statute, the Agency is entitled to use such evidence. It is, however, easy to dispute the veracity of intelligence data and potentially damaging for the IAEA to be seen to be too closely connected with national intelligence agencies. Such information would, therefore, often be best used as a 'tip-off' to

stimulate further investigation rather than as evidence in itself.

Analysis of open source data could also be useful. For instance, the absence of publications on a particular subject or by a particular scientist could indicate the existence of classified research. In addition, Agency investigations into a state's fuel cycle may uncover links to a weaponization programme. Iran's interest in polonium-210, for instance, was discovered when the Agency found that Iran had irradiated bismuth-209 in a reactor without declaring it [14].

| | Data source | Equipment | Effluent | External physical features | Other |
|---|---|---|---|---|---|
| **Inspections** | Visual observations | ● | | ● | ● |
| | Interviews with scientists and officials | ● | | | ● |
| | Environmental sampling (wide-area or site-specific) | | ● | | |
| | Plant stress | | ● | | |
| | Documentary evidence | ● | ● | ● | ● |
| **Remote monitoring** | High-resolution satellite imagery (national means or commercial) | | | ● | |
| | Hyperspectral satellite imagery | | ● | | |
| | Airborne air sampling | | ● | | |
| | Patterns of electricity use | ● | | | |
| **Export and financial data** | Export data (including refusals) | ● | | | |
| | International banking data | ● | | | ● |
| | Investigations into the black market | ● | | | ● |
| **Open source data** | Media reports | | | | ● |
| | Scientific literature | | | | ● |
| | Reports from foreign scientists | | | | ● |
| **Intelligence** | National means of intelligence | ● | ● | ● | ● |

**Table 2:** Verification techniques that could be useful for detecting clandestine weaponization activities. 'Other' indicates signatures and indicators (such as documentary evidence or changes in the structure of the scientific community) that are not covered by the headings 'equipment', 'effluent' or 'external physical features'.

It is also possible that initial evidence could be obtained from investigations into the nuclear black market, especially if equipment used in the weaponization process were being sold. If black market transactions only involved information (in the form of documents, say) then they would be much harder to detect during the early parts of an investigation.

In theory, remote monitoring could also be useful at an early stage. In practice, however, it would more likely become an important tool at a later stage—once suspect sites had been identified. Note that remote monitoring, like a number of the tools in table 2, is most effective at identifying the presence of industrial activity where there should be none. It is less effective at distinguishing between legitimate and illicit activities. For instance, satellite imagery showing construction work on a large underground facility in an isolated area would legitimately create suspicions. Similarly, the presence of high voltage power lines at a supposed agricultural facility would also be worth investigating further. In contrast, it would be much harder to use satellite imagery to detect a facility involved in weaponization if it were based near a number of existing industrial facilities.

Once a suspect site has been identified, environmental sampling would be useful in ascertaining its purpose. For detecting fuel cycle facilities it is probably sufficient for the

Agency to test for the presence of uranium and plutonium, their decay products and their isotopic composition. To identify clandestine weaponization activities, it would be useful for the Agency to attempt to detect a much broader range of non-nuclear materials, including those listed in table 1.

Ultimately, concrete evidence of a weaponization programme would probably only result from much more intrusive investigations. Inspections of suspect facilities, interviews with scientists and officials and access to documents would likely be required. It seems very unlikely that a state genuinely intent on developing nuclear weapons would be co-operative. This would severely complicate the task of conducting inspections and the Agency would likely require strong and active support from the UN Security Council.

### 3.4. Case study: The detection of hydrodynamic testing

Although a gun-type device is technically simpler, most proliferators have sought implosion weapons, which require less fissile material, can be miniaturized for use on a ballistic missile and can have higher yields. In an implosion device, conventional explosives are used to drive the pit into a supercritical state. Maintaining the symmetry of the pit during implosion and ensuring that it reaches a sufficiently high density before initiation are two of the toughest challenges facing a proliferator. To develop a theoretical understanding and test potential weapon designs, the behaviour of the pit during implosion is studied experimentally by replacing the fissile core with a non-fissile 'simulant' during a so-called hydrodynamic test [15]. Hydrodynamic tests are probably the most easily detectable weaponization activity. A discussion of them affords an opportunity to illustrate the process of detecting clandestine weaponization activities in more depth.

Monitoring the shape of the pit during implosion is a challenging task requiring complex, highly specialized equipment such as flash X-ray generators and recording systems (see table 1 for a fuller list) [16]. A state developing nuclear weapons may well decide against trying to develop such technology indigenously, in which case evidence of its intentions could come from attempts to procure relevant items of equipment or their components. It might also be possible to detect a state's efforts to build up a relevant body of expertise through analyzing open source literature.

Small-scale hydrodynamic tests might be conducted indoors at a specialized facility (although this is not necessary). A recent description of the Hydrodynamics Facility at Britain's Atomic Weapons Establishment gives a flavour of what such a facility might be like [17]—although a state developing its first nuclear weapon would certainly not need a facility as large or complex as the British one. Satellite imagery could be used to detect the internal structure of such a facility during construction. Satellite imagery could also be useful for detecting the outdoor facilities used for full-scale hydrodynamic testing. American facilities are known to consist of concrete pads with housing for equipment on either side [18]. It would be convenient, but not by any means necessary, to base hydrodynamics testing at a pre-existing ordinance facility.

The Agency might be able to detect evidence of hydrodynamic testing after the event by environmental sampling. Evidence obtained in this way would be particularly compelling if traces of simulant, tamper or reflector and high explosive were found in combination. For a high enriched uranium weapon, depleted or natural uranium would be an obvious choice for a simulant. It is known that lead, tantalum and depleted uranium have all been used to simulate plutonium devices (this list is almost certainly not exhaustive) [19]. Beryllium is believed to be a common choice for use as a reflector. Tungsten, depleted or natural uranium can be used as tampers. For a short period following a test, damage to the surrounding vegetation might also be visible. An onsite inspection would be required, however, to observe this.

Further evidence of hydrodynamic testing could come from interviews with people living near a test site (although it is very possible that tests would be carried out far from populated areas). Apart from the noise created by the explosion itself, a warning klaxon before a test might be sounded. Bright 'streamers' of molten uranium expelled from the pit would also be visible if natural or depleted uranium were used as a simulant [20]. It has also been suggested that once a test site has been identified, a satellite could be used to monitor it for possible hydrodynamics testing [21].

There is no doubt that more work is needed to determine the feasibility of detecting

hydrodynamic tests. In particular, the efficacy of using environmental sampling requires much more investigation. Nonetheless, hydrodynamic tests do leave a considerable footprint, and it is at least plausible to suggest that their detection is possible. Indeed, the Agency did uncover some evidence for hydrodynamic tests during its continuing investigations in Iran [22].

## 4. Conclusions

The difficulty of detecting clandestine weaponization activities should not be understated. The research and development associated with manufacturing a nuclear weapon is generally small-scale and hard to distinguish from many other types of legitimate scientific and industrial activity. There are, however, a number of characteristic indicators and signatures associated with a weaponization programme. Although a proliferator will likely try to minimize the footprint left by manufacturing a nuclear weapon it cannot be eliminated entirely. In particular, certain weaponization activities such as hydrodynamic tests present a realistic opportunity for detection.

Some idea of the feasibility of uncovering a clandestine weaponization programme comes from the Agency's continuing investigations in Iran. During these investigations, the IAEA has uncovered evidence which may point to efforts to manufacture nuclear weapons (such as the production of polonium-210 [23], possible hydrodynamic testing [24], and the existence of a document describing the machining and casting of uranium into hemispheres [25]). It did so using only the authority of its existing mandate and without access to a number of the verification techniques discussed in this paper. It is therefore plausible to suggest that, even if the Agency could never be in a position to provide credible assurances about the absence of weaponization activities in a state, systematic and concerted searches would have a reasonable chance of success.

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# Session 11

# CS-II

# C/S in Final Disposal Processes - Swedish and Finnish perspectives

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***Abstract:***

*Sweden and Finland are planning the same concept for the disposal of spent nuclear fuel: the once-through nuclear fuel cycle with the final disposal in geological repositories deep in the bedrock. Even though the fuel is planned to be retrievable, there will be a very high degree of inaccessibility of the nuclear fuel once it is emplaced in the repositories. This is a trait that is unique among the nuclear facilities that are under safeguards today. For the credibility of safeguards, the safeguards authorities must ensure that the fuel to be disposed of is verified and that the continuity of knowledge is securely maintained, since no re-verification is feasible if the continuity of knowledge should be lost. This makes a robust and reliable system of containment and surveillance very important. This paper identifies the boundary conditions of such a system, discusses the role of the safeguards authorities and suggests ways of implementation of containment and surveillance in the final disposal process of Finland and Sweden.*

**Keywords:** Containment and Surveillance; Final Disposal; Geological Repositories

## 1. Introduction: Final disposal in Finland and Sweden

Sweden and Finland are two countries with similar strategies within the field of nuclear power. For example, both Sweden and Finland implements the once through fuel cycle where no fuel is reprocessed. This strategy, and the fact that the Fennoscandian peninsula has stable crystalline bedrock, makes geological repositories the natural choice for the disposal of spent nuclear fuel. The similarities between the two countries make close cooperation not only possible but also clearly advantageous. The companies responsible for final disposal, Posiva in Finland and SKB in Sweden have had cooperative research and jointly funded projects for years.

Sweden and Finland plan to encapsulate the spent fuel in cast iron enforced copper canisters, which will be emplaced in the geological repository. Finland has decided on the repository site, while Sweden is in the phase of site selection with the location of the encapsulation plant already suggested.

The final disposal procedures in Sweden and Finland will eventually render the spent fuel unavailable for verification or identification. The degree of inaccessibility after disposal will be unique for geological repositories. Together with the exceptional time spans related to the final disposal, this trait presents an unprecedented challenge for designing a credible safeguards concept. An important consequence of the inaccessibility is that all safeguards issues must be resolved prior to emplacement and backfilling.

The disposal process will be of a continuous nature, with a number of fuel assemblies being prepared for encapsulation and final disposal each week. Since no similar operations take place in the nuclear power industry in Sweden or Finland today, the present safeguards system needs to be adjusted to handle this new situation. Although continuous processes like enrichment and reprocessing exist today in other countries, the methodologies there cannot be directly transferred onto the final disposal process. This is due to an important difference: the geological repository does not have an outflow that can be investigated to confirm the presence of all nuclear material. This feature makes it necessary to create a new safeguards approach for final disposal of spent nuclear fuel.

In Finland and Sweden alike, there is a common understanding that comprehensive NDA verification should be performed prior to encapsulation. Verification provides the operator, the safeguards authorities and future generations understanding of the properties of the disposed material. The continuity of knowledge (CoK) must be vigilantly maintained

after the NDA verification. Hence, a robust and reliable system of Containment and Surveillance (C/S) is of vital importance.

This paper presents C/S considerations based on the current views and plans for final disposal in Finland and Sweden.

## 2. The concept

The Swedish and Finnish concepts both consist of three blocks [1, 2]:

1. Wet interim storage,
2. Encapsulation plant and
3. Final disposal tunnels.

At present the spent fuel is stored in wet intermediate storages. These facilities will constitute the starting point of the final disposal process. The encapsulation facility is either adjacent to the interim storage, within the same nuclear site or further away, which makes fuel transport to the encapsulation facility necessary. In Finland the transport is inevitable, because Loviisa NPP fuel need to be transported to Olkiluoto, where the encapsulation facility and final disposal site is located. Sweden has one central intermediate storage facility, Clab in Oskarshamn, and the main alternative in Sweden, as presented by the operators, is that the encapsulation facility should be directly connected to Clab.

In the encapsulation facility the spent fuel assemblies will be placed in copper canisters, which are then welded shut. From a safeguards point of view the encapsulation is a re-batching process where 12 (BWR, VVER) or 4 (PWR) fuel assemblies (nuclear fuel items) are placed in one copper canister. The canister thus forms the new fuel item of interest for the safeguards system and must therefore have a batch ID that can be read manually or using adequate technical equipment. In this paper it is proposed that the new fuel item (the canister) is defined to be created immediately after the canister has been filled and sealed by the welding of the lid.

After encapsulation the spent fuel canisters will be transferred or transported to final disposal tunnels and emplaced in disposal holes, which will be backfilled. The operator wants to keep open rock volume as small as possible, so the excavation of new disposal tunnels and filling the old ones is an ongoing process together with final disposal operations.

## 3. The boundary conditions for the safeguards system

A few boundary conditions for the safeguards system have been identified:

- First of all, the system must create credible and accurate information about the disposed fuel. After disposal the fuel cannot be verified, therefore the conclusion must be clear, unambiguous, accepted by all parties and well documented. This requirement gives us guidance that:
  - all spent fuel should be verified at least on a partial defect level or in any case the best technology available. In the case that the best available measuring technique is deemed insufficient as a method for creating credible and accurate information, the system should be prepared, both technically and conceptually, for the incorporation of new, better technologies as they evolve.
  - after verification the CoK should be well maintained. In case of failure, the CoK must be re-established using an adequate method, for example NDA.
  - there should be a mechanism created, through which the relevant parties (operator, safeguards authorities, the IAEA) can give their statements and clearances before the material becomes difficult to access. This brings in a requirement of sufficient buffer capacity between the final verification and the encapsulation/disposal process.

- The safeguards concept should be feasible relating to operational resources. For instance, the system should not require constant physical presence by the safeguards authorities or by the IAEA. Nor should it interfere with the operator's regular activities, implying that the system and its' components should be robust and reliable. Sufficient amounts of spare parts and components should be made available for all relevant technical systems. For this point also, an extra buffer of fuel items ready for disposal is beneficial. The reason for this proposal is that the safeguards measures should not hamper the encapsulation and disposal processes in case of a failure in the NDA system. With a buffer, material ready for disposal will always be available.

- The present three-month or one year inspection interval is not a functional strategy for the final disposal process, since the material (according to plans) will be rendered difficult to access continuously. This makes verification of inventory impossible.

- Backflow of the material in the process chain could disturb operations badly. Therefore, it should not be required for safeguards reasons. This means it should be possible to handle any failure (technical malfunctions, loss of CoK etc.) in the safeguards system without having to create a backflow of material. In case of process disturbances the operator may need to move fuel backwards in the process. Such situations should be analysed on a case-by-case basis by a relevant party.

## 4. Authority requirements

The Swedish and Finnish safeguards authorities have not yet issued any requirements for the safeguards system for the final disposal processes. Under any circumstance these requirements cannot be less stringent than those (still undefined) of the IAEA. Some guidelines could be found in the results of the IAEA Consultants Group Meeting of 1995 [3]. The meeting recommended that "spent fuel will be disposed of only as verified nuclear material on which continuity of knowledge (CoK) has been maintained". In the safeguards terminology, CoK is considered maintained as long as the C/S system that has been applied on the nuclear material can be evaluated as acceptable. This implies that the C/S system needs to be very robust and reliable. A dual C/S system is therefore the reasonable choice and should be required from the authorities. In this context it may turn out to be feasible to re-define dual C/S in such a way that it requires two conclusive positive results in order for a conclusion of maintained CoK, "enhanced dual C/S".

On a national level the authorities concern is to provide future generations credible assurance that the nuclear material declared to be disposed of actually is emplaced in the declared location and with declared properties. This means that correct and complete records have to be maintained over centuries, which implies two challenges:

- to ensure that correct and complete documentation of all nuclear material is created, and
- to preserve this documentation so that it is accessible to future generations.

The authorities should thus require that only verified material on which continuity of knowledge has been kept, is allowed to pass through the disposal process.

The archival challenge must also be met, not only in the sense that the data and information must be available during an unforeseeable time-span but also considering authentication of the data.

An important question for the safeguards authorities to consider is of the ownership and maintenance of safeguards equipment: should the equipment belong to the authorities or the operators? When weighing these alternatives, the authorities will have to balance the integrity of the collected information against the intrusiveness on operations and resource efficiency. If the authorities own, operate and maintain for example the NDA equipment, it will be highly resource demanding since the equipment will be used continuously. Remote operation of the equipment poses large initial costs but could be more economical in the long run. However, in case of a failure, the operators have to halt operations until personnel from the authority arrive and repair the apparatus. On the other hand, the integrity of the implemented safeguards system could be questioned if the operators own, operate and maintain the measuring equipment. A related issue is if the IAEA could use information from the national authorities safeguards systems. These issues should be carefully considered by the authorities and the IAEA.

## 5. C/S functions

Possible diversion scenarios are described in SAGOR reports [4]. To cover the diversion scenarios we propose the following functions for the C/S of a final disposal safeguards system. It should be noted that C/S functions are not meant to be the only safeguards measures but are complemented with other methods.

1. The assurance of integrity and identity of the fuel elements after the final verification until encapsulation is achieved by constant surveillance and appropriate sealing of items. Dual C/S (or perhaps "enhanced"

dual C/S) is recommended, as losing CoK at this stage would interrupt the encapsulation process and may require re-verification. A fingerprinting system of fuel elements would make confirmation of CoK more robust and, consequently, requirements of the C/S systems could be relaxed. If this phase includes fuel transport over long distances, a tracking system may be appropriate.

2. The encapsulation process is performed under constant, uninterrupted surveillance. For monitoring the flow in the encapsulation facility, portal monitoring is a viable option.

3. The CoK must be maintained also in the tunnel system of the repository. A risk assessment analysis is necessary to find the best way to achieve this. Environmental conditions in the disposal tunnels excavated in the crystalline bedrock in Sweden or Finland are expected to be such as to allow for the use of portal monitoring systems or other forms of robust radioactivity measurements. Monitoring of only the access points to the repository and seeing the tunnel system as a "black box" is a desirable safeguards approach because of the relatively low costs for equipment and labour effort. However, a risk assessment may give the result that the "black box" approach is not sufficiently secure. In that case it should be complemented with additional safeguards activities underground. An example is that the emplacement vehicle could be equipped with a fingerprint reader and a positioning and radiation monitoring system to detect any replacement attempts. After the canister is emplaced, non-C/S activities like inspections and environmental sampling could reveal any undeclared canister opening activities. For the purpose of monitoring activities in the final disposal tunnels, the performance of portal monitoring should be investigated. It should be sensitive enough to detect the diversion of a shielded canister to be of use for safeguards purposes.

## 6. Process models

The requirements and the performance of a safeguards system can be analysed only if the disposal process is known. At the moment the process is not definitely defined either in Finland or in Sweden. Since the Finnish plans are more evolved, this preliminary model is created from the Finnish case.

### 6.1 Analysis of Process Model "FIN-1"

The first safeguards concept for Finnish final disposal process is proposed in a STUK report [5]. The assumptions of the concept are:

- The model envisions two interim storages, one of them being within the same site as the planned encapsulation plant and geological repository.
- Final disposal tunnels are within the same site as the encapsulation plant.
- Encapsulation is an ongoing process with a capacity of 2 canisters/week.
- Final NDA verification will be performed at the interim storages, which have separate pool areas where verified fuel will be stored. These pools serve as a buffer zone.

Let us use this concept as a basis of our first process model and call it "FIN-1". The early NDA verification is a benefit because of large buffer storages of verified fuel ready to be introduced into the final disposal process, and also longer analysis periods. There is time to draw safeguards conclusions about the fuel after the NDA measurement. This makes the system non-intrusive.

After the NDA verification the challenge is to maintain the CoK. We propose to develop special fingerprinting methods, which would be used simultaneously with the NDA verification. A "fingerprint" would be a specific pattern based on a unique physical characteristic of the fuel assembly. This pattern should be practically impossible to counterfeit. An example of such a method is reading the microstructures of the fuel item surfaces [6]. The selected method could be combined with weighing and radiation pattern measurements of the fuel assembly to detect if the integrity of the item has been breached.

The fingerprinting method would make the C/S system more robust since the fingerprint would provide identification of the item and evidence that the item integrity is intact. However, no such system is in operational use today. The development of such a method is however desirable. Potential candidates exist, like surface scanning methods, which can be used to uniquely identify surfaces or bolts of the fuel assembly. Replacement of an assembly or removing the bolts would be detected. A

similar fingerprinting method could be also used for the copper canister to verify its identity and integrity.

All buffer storages and fuel movement operations should be continuously monitored with a camera. Sealing may be used as a complementary method. For fuel transports a tracking system is proposed.

Underground operations are challenging to monitor. If all outlets of the tunnels can be comprehensively portal monitored it could be a backbone of the safeguards system. However, the detection capability of portal monitors should be investigated. The need of other C/S methods or even use of novel technologies can be analysed, when the process is better known.

The safeguards processes of FIN-1 model are described in Table 1. Some of the proposed methods are optional and further analysis is definitely needed.

| **Process** | **Purpose** | **Technical method** | **Operational requirement** |
|---|---|---|---|
| NDA verification | -To verify the correctness and completeness of operators records | Tomography or another partial defect method | Reliability: medium (MTBF = few years).<br>Servicing time less than 1 month<br>A spare system may be required<br>Remote monitored |
| Fingerprinting the assembly | To create unambiguous fingerprint from the assembly for later identification. The fingerprinting and NDA verification are simultaneous operations. Fingerprinting should also detect if the integrity of the assembly is touched | Surface fingerprinting (or some other identification method)<br><br>Radiation fingerprinting<br><br>Weighing<br><br>Further research required | Reliability: medium (MTBF = few years).<br>Probability of false positives or false negatives lower than $10^{-6}$. Servicing time less than 1 month<br>A spare system may be required<br>Remote monitored<br>Integration with NDA verifier (fingerprinting and verification are simultaneous operations) |
| Buffer storing of verified assemblies | To maintain CoK during the cask loading | Camera surveillance | Camera reliability medium (MTBF = few years).<br>Remote monitoring desirable. |
| Transport cask loading | To maintain CoK during the cask loading | Authenticated fingerprint maintains CoK.<br>Camera surveillance | Camera reliability medium (MTBF = few years).<br>Remote monitoring desirable. |
| Cask transport/transfer | To maintain CoK during the transport, detection of covert opening and replacement | Authenticated fingerprint maintains CoK<br>Seals | Seal reliability high (MTBF = several years).<br>Remote monitoring desirable. |
| Cask storing | To maintain CoK during the cask loading,<br>detection of covert opening and replacement | Authenticated fingerprint maintains CoK,<br>Seals | Seal reliability high (MTBF = several years).<br>Remote monitoring desirable. |
| Cask unload | To maintain CoK during the cask loading, detection of | Authenticated fingerprint maintains | Camera reliability high (MTBF = several |

| | | | |
|---|---|---|---|
| | diversion and substitution | CoK<br>Camera surveillance | years).<br>Remote monitoring essential. |
| Encapsulation process | To verify CoK of the assemblies | Verification of fingerprints. remote monitoring Cameras, inspector presence, portal monitors. All diversion routes should be covered. | Overall reliability: high (MTBF = several years).<br>Servicing time: few days<br>Remote monitoring system required |
| Fingerprinting the canister | To create unambiguous fingerprint from the canister for later identification. Fingerprinting should also detect if the integrity of the canister is touched | Surface fingerprinting (or some other identification method)<br><br>or some other novel method.<br><br>Radiation fingerprinting<br><br>Weighing<br><br>Further research required | Overall reliability: high (MTBF = several years).<br>Servicing time: few days<br>Remote monitoring system required.<br>Probability of false positives or false negatives lower than $10^{-6}$. |
| Canister transfer to the emplacement hole | To maintain CoK during the transfer | Authenticated fingerprint maintains CoK<br>Portal monitors. All diversion routes should be covered. | Overall reliability: high (MTBF = several years).<br>Servicing time: few days<br>Remote monitoring system required |
| Emplacement of the canister | To verify CoK of the canister | Verification of fingerprints. Remote monitoring Cameras, inspector presence, portal monitors. All diversion routes should be covered. | Overall reliability: high (MTBF = several years).<br>Servicing time: few days<br>Remote monitoring system required |
| Backfilling | To detect undeclared removal of the canister | Cameras, inspector presence, portal monitors. Novel methodologies | Overall reliability: extremely high (MTBF = tens of years, Dual systems & redundancy).<br>Servicing time: few days |
| After backfilling | To detect undeclared access to the repository/canister. | Sealing, geophysical methods, satellite monitoring. Novel methodologies | Passive systems desirable. |

*Table 1. C/S and other safeguards methods in final disposal process model "FIN-1".*

## 7. Challenges

This paper has identified a number of challenges that can be resolved either by research efforts or by analyses and decisions made by the safeguards authorities:

- An NDA verification system, which would satisfy the needs for partial defect level verification does not exist today. Development of a tomographic verifying technique could potentially solve this problem. The work is done under the IAEA Support Program task JNT 1510, participated actively by Finland and Sweden.

- The performance of portal monitors for use in the final disposal tunnels needs investigation. If portal monitors could be made sensitive enough to detect the diversion of a completely shielded canister they could be implemented with the purpose of monitoring activities in the repository.

- A fingerprinting method to be used on fuel assemblies before encapsulation, and on the closed disposal canisters after encapsulation is desirable. The method should be fast and with a very low probability of false identification.

- For a resource efficient operation of safeguards equipment, remote monitoring will be needed. This concept will have to be developed further, including extensive reliability testing.

- A lot of effort (see for example [7])is being put into finding a secure way of transferring information to future generations. This research should be closely watched by the national safeguards authorities, whose task it is to inform the coming generations of the contents of the geological repositories. A method to authenticate the information to be placed in archives would also be welcome, since the correctness of the information is as important as its' completeness.

- The question of ownership of safeguards equipment should be resolved. The IAEA also need to decide on the possibility of using information created by the national safeguards authorities.

- As has been stressed above, the C/S system needs to be both robust and reliable. If the C/S system were to deliver inconclusive results the operation of the final disposal process will be affected negatively. A methodology of determining the C/S system's ability to deliver conclusive results is therefore desirable. At Uppsala University in Sweden, a research project is planned, aiming at quantifying the ability of a C/S system to detect an anomaly and to reveal the course of actions leading to the anomaly. Such quantification would provide a possibility to logically connect the important concept of CoK to measurable entities. Furthermore, in the ESARDA Working Group on Containment and Surveillance a qualitative method of determining a system's Assurance and Performance is being developed.

- In Sweden and Finland today, the safeguards system does not operate on continuous processes like the future final disposal process. This means that the methods used today are not well adapted to the future situation when the encapsulation plant and the repository are in operation. Some methods used in safeguarding continuous processes abroad, like enrichment or reprocessing, could possibly be implemented but there is one very important difference: The enrichment- and reprocessing plants have an out-flow that could be compared to the in-flow to verify that all nuclear material is accounted for, while the material emplaced in the repository will not be accessible for verification. This is a completely new trait, unique among all nuclear facilities in operation today, and it will demand a concentrated effort from all instances involved in safeguards in Sweden and Finland. This means that close co-operation, including the authorities SKI (Sweden) and STUK (Finland) will be of great benefit for both countries.

- Connected to the previous item is the issue of measuring strategy, measuring accuracy and confidence levels. To obtain a safeguards system with predictable properties and as cost effective as possible, it seems important to define exactly what should be measured, and for what purpose. The measurements should thus be performed with rationally defined limits on accuracies in order to be able to quantify the performance of the implemented system as a whole. This is important in order to be able to determine

with which confidence one may state that the processes run according to regulations and declarations. Hence, we propose that a project addressing these issues is initiated.

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# Test Procedures for the Radiation Qualification of Safeguards Systems Operating in Nuclear Radiation Fields

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***Abstract:***

*Safeguards equipment, in particular, electronic systems designated for unattended operation, has to be designed and approved for failure-free operation under different environmental conditions, e.g. nuclear or ionising radiation. Irradiation testing has to be performed in order to ensure the safe and reliable use as well as long-term survivability of electronic safeguards equipment in radiation environments typically occurring in nuclear installations. If radiation survivability measures are established already in the design stage of an electronic module when appropriate components such as microprocessors are being selected the time delay and additional costs will be negligible. Hitherto, the International Atomic Energy Agency (IAEA) and the Euratom Safeguards Authority were missing agreed test procedures and criteria for the qualification of safeguards equipment. Therefore, guidelines were developed for the application of universal measurement methods in irradiation tests.*

*The radiation environments were adopted as defined in the draft version 2.04 of the proposal "Common Qualification Test Criteria for New Safeguards Equipment". These criteria have been worked out jointly by IAEA and Euratom. Depending on the radiation level the tested equipment is categorised into three classes: standard, high class and special application.*

*The paper addresses the physical background of permanent damage and temporary malfunction of electronic or photonic systems, the guidelines for nuclear radiation testing of safeguards equipment, the criteria applied, and the statistical methods to be used. Test techniques have been defined for irradiation testing and parts control which, if applied, provide assurance that each delivered system will meet the radiation requirements. Guidance is given for managing the tests and templates for each sort of irradiation test. A set of templates was designed for the complete test documentation. The present document draws upon existing test and measurement standards. The described test methods are independent of any specific safeguards system. These test procedures and criteria have to be reviewed and, if necessary, revised according to the experience gained.*

**Keywords:** radiation effects in electronic; test of safeguards instrumentation; radiation hardness assurance; nuclear radiation environment; nuclear radiation testing

## 1. Introduction

The purpose of the intended document [1] is to establish universal measurement procedures for qualification testing, in order to ensure the safe and reliable use as well as long-term survivability of electronic safeguards equipment in radiation environments typically occurring in nuclear installations.

These test procedures are independent of any specific safeguards system. It is self-evident that these procedures have to be adjusted and further developed in the future according to the experience gained.

The International Atomic Energy Agency (IAEA) uses unattended safeguards equipment in nuclear

facilities worldwide. The equipment has to be designed for failure-free operation under different radiation conditions. Therefore, procedures have been defined for radiation testing and parts control which, if applied, provide assurance that each delivered system will meet the radiation requirements. The present document draws upon existing test and measurement standards.

By means of the proposed procedures the originator who is responsible for initiating the radiation test and the operator of the test facility will be able to prepare, perform, and evaluate appropriate qualification tests of the device under investigation. For all necessary steps templates are given in the document as well as a guidance to fill in the templates. Moreover, the explanations may assist the manufacturer in designing and producing systems that will have to operate for a long time in nuclear radiation environments.

The present paper addresses permanent damage and temporary malfunction of electronic or photonic systems resulting from total ionising radiation dose, neutron displacement damage, or neutron-induced single event effects. While the test procedures laid down in the report represent the current state of knowledge, they will have to be adapted if the application of future devices would bring up new effects that have to be consired.

## 2. Radiation Environment

The adopted radiation environments are defined in the draft version 2.04 of the proposal "Common Qualification Test Criteria for New Safeguards Equipment" [2]. These test criteria have been worked out jointly by IAEA and Euratom. The equipment is categorised into three classes: Standard = level 1, high class = level 2, and special application = level 3. The specified values are given in Table 1.

| Equipment Class = Test Level | Standard = 1 | High Class = 2 | Special Applic. = 3 |
|---|---|---|---|
| corresponding values for levels [1] | 0.1 mSv/h for 7 y | 5 mSv/h for 7 y | to be defined |
| total dose equivalent ($\gamma$+n) | 6 Sv | 300 Sv | 3000 Sv [2] |
| gamma dose (50 % of total dose) [3] | 3 Gy | 150 Gy | 1500 Gy [2] |
| gamma dose rate (recomm.) [4] | 0.2 Gy/h | 3 Gy/h | 20 Gy/h |
| irradiation time (dose/dose rate) | 15 h | 50 h | 75 h |
| thermal neutron fluence (25 %) [3] | $1.5 \cdot 10^{11}$ n/cm$^2$ | $7.5 \cdot 10^{12}$ n/cm$^2$ | $7.5 \cdot 10^{13}$ n/cm$^2$ |
| thermal neutron flux (recomm.) [4] | $1 \cdot 10^{7}$ n/cm$^2$s | $1 \cdot 10^{8}$ n/cm$^2$s | $5 \cdot 10^{8}$ n/cm$^2$s |
| irradiation time (fluence/flux) | 4 h 10 min | 20 h 50 min | 41 h 40 min |
| fast neutron (2 MeV) fluence [3] [5] | $3.7 \cdot 10^{9}$ n/cm$^2$ | $1.8 \cdot 10^{11}$ n/cm$^2$ | $1.8 \cdot 10^{12}$ n/cm$^2$ |
| fast neutron (2 MeV) flux [4] | $2 \cdot 10^{5}$ n/cm$^2$s | $2 \cdot 10^{6}$ n/cm$^2$s | $5 \cdot 10^{6}$ n/cm$^2$s |
| irradiation time (fluence/flux) | 5 h 10 min | 25 h | 100 h |
| fast neutron (14 MeV) fluence [3] [5] | $3 \cdot 10^{9}$ n/cm$^2$ | $1.5 \cdot 10^{11}$ n/cm$^2$ | $1.5 \cdot 10^{12}$ n/cm$^2$ |
| fast neutron (14 MeV) flux [4] | $2 \cdot 10^{5}$ n/cm$^2$s | $2 \cdot 10^{6}$ n/cm$^2$s | $5 \cdot 10^{6}$ n/cm$^2$s |
| irradiation time (fluence/flux) | 4 h 10 min | 20 h 50 min | 83 h 20 min |

**Table 1:** Radiation levels for test procedure from [2]. The irradiation values to be applied are highlighted.

According to current IAEA requirements safeguards equipment categorised “standard” and “high class” shall be used for 7 years. The dose values for “standard” and “high class” were defined by IAEA and Euratom, whereas for the “special application”-category there is no guidance, i.e., the values have to be defined on a case-by-case basis. Therefore the values for “special application” given in Table 1 are examples only. The proposed values for the test levels should be adjusted if changes in the specified radiation environments become known to the Agency.

## 3. Radiation Effects Testing

There are two types of effects of nuclear radiation on electronic components in a system: cumulative damage and transient effects. This distinction, however, is not always clear-cut. Damages may not be permanent, since the radiation-induced degradation is often reversed through annealing, and transient effects may result in permanent damage. Single event effects (SEE) may or may not result in permanent damage. They are caused by the passage of neutrons with the subsequent production of ionising particles. These in turn deposit electrical charges in a circuit node. The result will be a change of state or a transient pulse.

Prolonged exposure to energetic radiation can cause fundamental changes in the operating parameters and/or functionality of integrated circuits, sensors, power systems, charge-coupled devices (CCD), microprocessors, communication or interface subsystems, and other devices. The changes are typically irreversible. Suitable operating parameters can and should be monitored for state-of-health. While some of the radiation-induced defects that cause device degradation can anneal slowly over time, the damage rate may dominate the annealing rate during times of intense radiation, resulting in a net degradation. Permanent device degradation results from one of the following mechanisms: Total ionising dose effects (TID), displacement damage effects (DD), or single event effects (SEE).

The external radiation environment consists of different radiation types spanning a wide range in energy and flux. The environment inside the system box or housing can be significantly different from the free-field environment as a result of the interaction of the incident particles and photons with surrounding materials. Because of the broad spectrum of particles and energies, it is difficult to simulate the actual radiation fields in the laboratory. To simplify matters, different degradation mechanisms or failure modes are treated independently of each other and investigated separately at suitable irradiation facilities.

The radiation effects of concern in the field are:

- Total Ionising Dose (TID) (caused by all different radiation types together during the whole lifetime),
- Displacement Damage (DD) (structural damage of semiconductors, mainly by neutrons),
- Single Event Effects (SEE) (due to neutrons or secondary particles and including both ‘soft errors’ and ‘hard errors’).

Although the irradiation facilities used to simulate the different environments are a major factor in radiation testing, further critical factors arise from the operating conditions of the device. The failure mechanisms of many electronic devices exposed to radiation are a strong function of operating bias, operating mode (off, standby or active), and temperature. The storage, operational, and irradiation histories may also be relevant. Ideally, devices should be tested under actual field conditions.

### 3.1. Total Ionising Dose Effects

The ionising radiation addressed here consists of gamma rays, electrons, and X-rays. The basic effect of the exposure of microelectronic circuits to ionising radiation is the generation of electron-hole pairs in the various materials of the device. The two types of device that are most seriously degraded by ionising radiation are metal oxide semiconductors (MOS) and silicon bipolar devices. The electrical characteristics of MOS devices are altered by radiation-induced trapped positive charges in the gate and/or field oxides, and by interface states that are formed in the interface between Si and $SiO_2$ (see Figure 1).

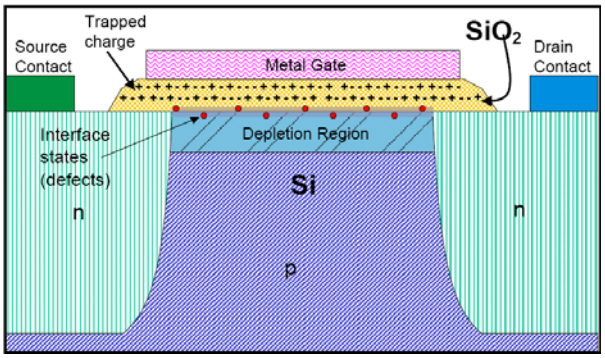


**Figure 1:** Mechanisms of total ionising dose damage in a MOS transistor

Ionising radiation generates defects in the insulating layers used in bipolar transistors and circuits. These defects can have serious effects on transistor operation. Radiation-induced positive charges in the oxides of bipolar transistors can cause increases in the leakage currents inherent in such devices. The effect of ionising radiation on bipolar transistors is the build-up of defects in the oxide covering the emitter-base junction of a substrate or lateral bipolar transistor (see Figure 2). These defects result in an increase of the base current and decrease of the collector current,

which in turn result in severe degradation of the transistor gain.

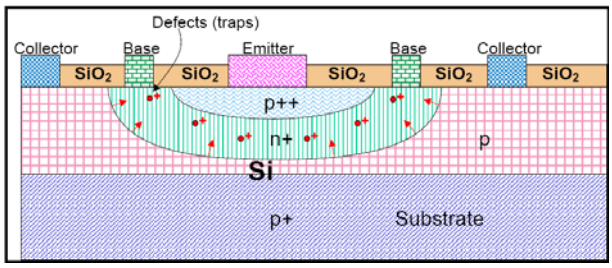


**Figure 2:** Mechanism of total ionising dose damage in a pnp lateral bipolar transistor

For TID, electronic performance degradation is caused by the absorption of ionising energy in sensitive material. The dose unit is Gray (1 Gy = 1 J/kg). Only the total value of TID over the lifetime is generally quoted as a radiation specification. TID testing is performed by exposing an electronic device to a well-calibrated radiation field while measuring the change in electrical performance.

The response of a device to radiation can depend on the test parameters. All such parameters must be defined and documented. These include: total applied dose, dose rate, operating conditions, electrical parameters, temperature, radiation source, dosimetry, annealing testing for metal oxide semiconductor (MOS), sample size, lot, vendor, statistical methods (confidence levels), and other experimental conditions.

The choice of an optimal procedure for performing a TID test includes the handling of the following three competing requirements.

- **Test fidelity**. It is necessary for a test to reproduce the results to be expected in the actual mission to an acceptable degree of precision. The chosen test methodology usually has a strong impact on the precision of the result. Often, it is necessary to have a good knowledge of the device physics, in order to perform laboratory tests that adequately match the device response to be expected in the actual use or to enable sufficiently correct extrapolation from laboratory results to in-field conditions.

- **Reproducibility**. It is important that the test results are reproducible in different laboratories. In order to reach this goal, careful attention must be paid to the control of experimental variables and to the use of accurate dosimetry methods.

- **Testability**. Testing must be economical in terms of time, equipment, and personnel. The "perfect" test will be too expensive to perform. The goal is to optimally balance financial expenses, volume of data, reliability of results, and available time.

Radioactive sources that are typically used for TID characterization include $^{60}$Co and $^{137}$Cs gamma sources. The most commonly used source for TID effects testing is $^{60}$Co. It emits gamma photons with energies of 1.17 MeV and 1.33 MeV (often only the mean of 1.25 MeV is specified) and has a half-life of 5.27 years. When using $^{60}$Co sources the device under test must be enclosed in an aluminium (or similar) container to minimise any dose enhancements caused by low-energy scattered radiation. Radiation sources based on $^{137}$Cs can be used for TID testing in much the same way as $^{60}$Co sources. The principal gamma ray energy is 662 keV. Some laboratories use spent fuel assemblies for TID testing. A typical gamma spectrum of nuclear fuel rods with long cooling time is shown in Figure 3.

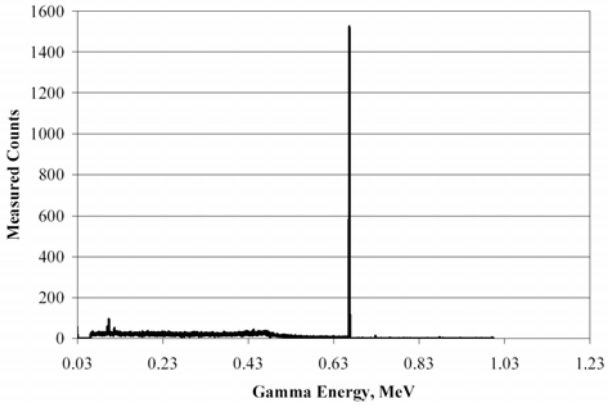


**Figure 3:** Measured gamma energy spectrum of nuclear fuel rods with 25 years cooling time. The $^{137}$Cs gamma line is clearly visible at 0.662 MeV. The low-energy photon tail due to Compton scattering is also shown [3].

The accurate determination of the absorbed dose in a semiconductor device requires knowledge of the elemental composition and geometrical structure of the test sample, the appropriate (tabulated) mass energy absorption coefficients, the energy spectrum of the incident radiation field, and a related measurement of the dose rate based on a dosimeter that has been calibrated according to national standards. Ionisation chambers, calorimeters and thermo luminescent dosimeters (TLD) are the preferred devices.

### 3.2. Neutron Induced Displacement Damage Effects

Neutrons interacting with semiconductor materials can produce modifications of the crystalline structure due to non-ionising energy loss (NIEL) displacement effects. These modifications can vary from vacancies (missing lattice atoms) and interstitials (atoms in intermediate, non-normal lattice locations) to damage clusters. Displacement damage can result in an increase of dark current and charge transfer inefficiency for CCDs, loss of gain in photonic and bipolar devices, and cell damage in DRAMs. Shielding, in general, does not reduce displacement damage effects as much as it reduces total dose effects.

As the displacement damage is mostly stable at ambient temperatures, no extrapolation is required to account for the rate at which the damage is introduced. Since the damage from neutrons is a strong function of energy (see Figure 4), the particle fluence versus energy must be integrated with the damage versus energy curve, in order to establish a damage equivalent fluence.

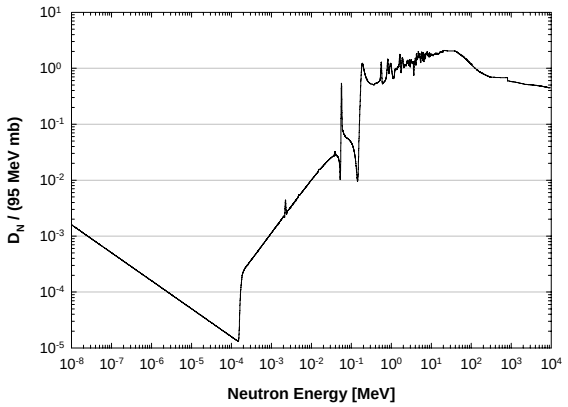


**Figure 4:** Displacement damage cross section $D_N$ of neutrons in silicon normalized to the damage value (95 MeV mb) of 1 MeV neutrons [4].

Material test reactors with beam tubes or hydraulic tubes or having a swimming-pool design, can be used for the exposure of samples to fast neutrons. The fast-neutron fluxes in the reactor core are accompanied by gamma rays from nuclear disintegrations with dose rates in the order of 10 kSv/h (1 MRad/h), which complicates the interpretation of responses for some modern CMOS devices.

Fusion-neutron sources are characterized by an almost discrete, mono-energetic neutron energy spectrum. An example is that of 14 MeV neutrons produced from the deuterium-tritium (D-T) nuclear fusion reaction. Typical commercial generators can emit up to a few times $10^{10}$ n/s in $4\pi$ or can produce pulses with a peak neutron flux of about $10^6$ cm$^{-2}$ s$^{-1}$ and a 0.1 µs half-width.

### 3.3. Neutron Induced Single Event Effects

Single Event Effects - SEEs include any kind of changes to an electronic device caused by a single particle strike. The radiation type of concern for SEEs are neutrons from nuclear fissions and accompanying secondary particles. Two trends support this increasing probability of occurrence of SEEs, namely the increased sensitivity of individual MOS transistors to upsets due to smaller feature sizes and lower supply voltages and the higher packing density of transistors. The aim of SEE tests is to measure the number of SEEs per unit fluence for that specific effect. The test results may be used to estimate the SEE rate or MTBF in the field via the ratio of the fluxes.

Many factors influence SEE testing. Hence, the test conditions have to be established beforehand (and recorded during the test) to ensure the validity of the test. Important factors that have to be considered when planning a SEE test are listed below.

- Type of SEE being tested;
- review of existing radiation test data for the device;
- energy of the neutrons;
- neutron flux; this must be low enough to avoid collective effects (pile-up);
- total particle fluence; the minimum fluence is that required to establish a statistically significant number of SEEs at any energy. The maximum fluence should be kept below that corresponding to a significant displacement damage degradation;
- physical and experimental set-up, which includes the particle source, dosimetry, cabling, vacuum chamber;
- geometry of the device under test with respect to the radiation source, e.g., the angle of incidence;
- test procedure, electrical biases, input vectors, temperature, current limits, clock rates, reset conditions, and others;
- prior history of TID; the SEE testing will deliver differing data if performed with new parts or with parts that are “pre-dosed” (e.g., during TID test);
- history of device and subsystem storage and operation, including burn-in.

## 4. Radiation Safety Aspects

During neutron irradiation nuclear reactions occur in the irradiated components (i.e., neutron activation). Therefore, the neutron-irradiated parts may be radioactive. Handling and storage of test specimens or equipment subjected to radiation environments have to be done with care and in consideration of regulatory limitations. Most countries have their own radiation protection regulations which have to be taken into account when working in controlled areas.

The radiation safety analysis has to focus on components having either a large cross section for nuclear reactions and/or a high mass combined with suitable half-lives of the product nuclei. In principle, the tested system has to be analysed with respect to the implemented chemical elements, nuclear isotopes, their mass values, and corresponding nuclear data.

In general, devices have at least three main components: housing, electronic board, and battery. Housings typically are made of plastic, aluminium or stainless steel. The activation during thermal neutron irradiation is larger compared to 2 MeV or 14 MeV neutron irradiation. Therefore, especially devices irradiated with thermal neutrons shall be handled with care. For mobile devices lithium batteries are increasingly used as power supply. Tritium will be created during thermal neutron irradiation. In the literature evidence can be found, that no safety risk exists concerning tritium gases in lithium batteries [5]. Electronic devices consist of many different elements; therefore, no general information can be given.

To avoid a safety risk due to high radioactivity, any exposed device has to be measured, before it is removed from the controlled area. The measurement should be made with a calibrated dose rate meter. From the radiation safety point of view there is a difference between the radiation testing and in-field usage because radiation testing involves a higher neutron flux.

## 5. Hardness Assurance Aspects

The fundamental challenge of any hardness assurance programme is to ensure in a cost-effective and timely manner that devices and systems will perform as intended in the anticipated radiation environments. As irradiation testing is usually considered destructive in the sense that the tested device is no longer suitable for use, nearly all hardness assurance testing depends on the sample size, i.e., number of devices undergoing testing. From this group of devices one must be able to derive, with a suitable confidence, the radiation response of the untested portion of the lot or batch that will be used in the field. This requires that the different sample devices must be relatively homogeneous. As nominally identical devices can show quite different radiation responses, special care must be taken to control the lot or batch.

There are many ways to ensure that the system will survive in the radiation environment and will work as intended. For example, one can use only radiation hard components or ones that have a well-known history of using in radiation environments. Or, if weight and space are of no concern, then shielding may guarantee the performance. In many cases redundancy, error detection and correction, or circumvention will help to mitigate system failures. However, any method used to accomplish the task has to be proven by irradiating the system in the laboratory prior to its in-field application.

Availability, variety, and cost, can enforce the deployment of Commercial-Off-The-Shelf (COTS) components rather than radiation hard parts. Radiation hardness is not a specification of COTS components. For COTS devices, the variation in the radiation sensitivity among components may be large [6] and can be exacerbated by the use of plastic packaging.

Electronic components often do not have sufficient radiation hardness to survive the doses and fluences expected during in-field use. In these cases, radiation shielding may be applied, so that parts are subjected to less radiation than in the unshielded case.

With increasing complexity of electronic systems, their reliability can be expected to decrease. One of the techniques for coping with this situation is to use corresponding electronics in redundant configurations. In principle, two approaches are possible: (1) Duplicating the basic components (or subsystems) in a complex system or (2) duplicating the complete system. In Figure 5 a comparison is shown for the two cases of redundancy.

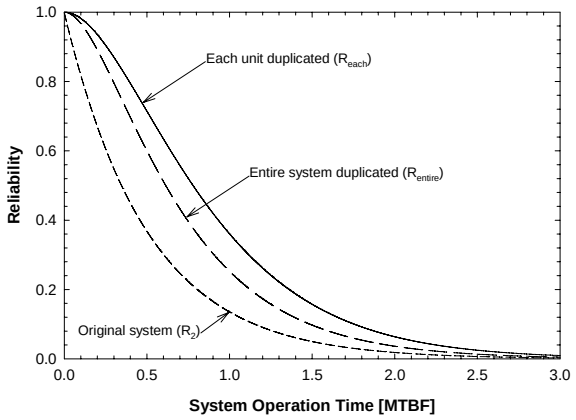


**Figure 5:** Reliability for "high-" and "low-level redundancy" of a system consisting of two units, each having a failure rate of $\lambda$ or a MTBF of $\lambda^{-1}$.

Circumvention is defined as the electronic process, where the system is transferred to a temporary state where all input and output ports are clamped to ground. Circumvention is used when the radiation levels exceed the normal levels many times and approaches the logic upset level, so that the system is more or less in the state of handling soft errors. Circumvention is usually accomplished through the use of a radiation detector that detects the increase of the incident radiation, an inhibit and recovery system, and a non-volatile memory where the actual status, programme, and data can be stored during the shut-down interval.

In some cases a reset or restart of an electronic system after its temporary malfunction for various reasons may restore its normal operation.

The radiation testing and qualification is the verification portion of the hardness assurance programme. The main objective is to provide direct experimental verification that procedural methods, including screens and controls used during the production process, are adequate to assure that previously established design goals of reliability are reached. Various levels of assembly should also be tested, and the selection of the appropriate level is based on several factors such as costs, timeliness, destructive or non-destructive tests, and the determination of the optimal level. For each radiation effect that has an impact on system performance, there is a specific level of assembly to be tested and type of measurement to be made before, during, and after irradiation. As all performed irradiations are accelerated tests (dose rates or fluxes are far above actual values in the field) annealing may enable devices, which failed directly after the end of the irradiation, to operate correctly again after some waiting time. This indicates that they may survive the dose of a lower test level or of the given test level if irradiated with lower dose rates or fluxes.

A precondition for the radiation qualification is to define pass/fail criteria. For instance, pass/fail criteria may concern electrical parameters such as power consumption that may change during testing. Definition of pass/fail criteria may result in the requirement of dedicated data acquisition equipment and thorough understanding of the device to be tested.

The radiation test report documents the procedures and results of tests and analysis performed, in order to establish the adequacy of the hardening design. All data and results are incorporated in the test report.

## 6. Radiation Hardness Assurance Documentation

The radiation hardness assurance documentation consists of a set of nine documents:

- Qualification Test Result Document

  In the Qualification Test Result Document, the overall result of the qualification test is given.

- Decision Tree for the Originator – Preparatory Actions
- Decision Tree for the Originator – Test and Evaluation

  The originator at IAEA will be guided through the procedure by using the decision trees for Preparatory Actions and Test and Evaluation.

- Pass/Fail Criteria

  The pass/fail criteria for single devices have to be defined by parameter, specified values, and limits. Additionally, for lot acceptance, pass/fail criteria must be given by choosing the number of devices to be tested and the number of devices allowed to fail.

- IAEA $\gamma$-Irradiation Test Document
- IAEA Thermal Neutron Irradiation Test Document
- IAEA Fast Neutron Irradiation Test Document

  For each type of radiation test and each device, separate IAEA irradiation test documents have to be used. The tests will be performed for each device step by step beginning with the $\gamma$-irradiation, followed by thermal neutrons, and, finally, fast neutrons.

During the test, the operator at the test facility and the equipment operator complete the Irradiation Test Documents. All irradiation tests should be performed with the hardware that is from the same batch/lot as the devices that will be used in the field. They must be operated under real working conditions.

- Radiation Safety Document

  Radiation safety aspects include a dose rate measurement of the neutron irradiated device after irradiation.

- System Data Sheet

  The System Data Sheet template has to be filled in by the manufacturer and delivered together with the devices to be tested.

## 7. Acknowledgements

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# How to Determine the Performance and Assurance of C/S Equipment

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**Abstract**

The issue of performance and assurance assessment of Containment and Surveillance (C/S) equipment is important for the development of an appropriate safeguards approach and for the reconsideration of existing approaches with regard to the implementation of the Additional Protocol and Integrated Safeguards. It is expected that the selection process of appropriate equipment for unattended operation is facilitated by the availability of methods to determine the performance and assurance of such equipment. The issue is being addressed by the ESARDA Working Group on (C/S) with top priority.

Apart from Euratom, the users of assessment methodologies would be the IAEA, plant operators, and instrument developers.

Equipment performance aims at the creation of relevant data, whereas assurance aims at the creation of information in support of the inspector's decision process.

In a first project phase the long-term dry storage of spent fuel assemblies is addressed. A methodological approach for determining the performance of C/S equipment includes the following components: (1) Information about the design and operational characteristics of the facility, (2) assumptions on diversion and misuse scenarios, (3) definition of safeguards requirements, (4) assessment on how the C/S equipment is suited to meet the safeguards requirements.

The principle to determine the performance of C/S equipment will be to define a task profile and to check the performance profile against the task profile. For instance, the task with respect to a cask loaded with spent nuclear fuel is to verify its identity and integrity. This can be achieved by applying a seal. The performance profile requires, among others, a sufficient radiation tolerance of the seal.

The paper will describe the development status of the methodological approach for performance and assurance assessment and highlight the input from the Working Group.

**Keywords:** containment and surveillance, equipment, performance, assurance, methodology

## Introduction

Material accountancy is a fundamental measure in the application of International Atomic Energy Agency (IAEA) safeguards, and containment and surveillance (C/S) measures are applied to complement material accountancy. Within a safeguards approach, two roles can be attributed to C/S equipment:

- to facilitate accountancy data acquisition;

Measurement data are "frozen" so that their validity can be carried forward into the future without re-measurement, e. g., by applying a seal on a spent fuel cask.

- to ensure that all material flows pass through key measurement points as declared.

This implies that all credible removal routes for material at the boundaries of a material balance area are covered by C/S devices; e. g., by applying surveillance to an open core in a reactor during refuelling.

The performance of safeguards measures can be determined in a quantitative way with commonly recognised mathematical methods, as long as the measures involved only consist of quantitative measurements with known measurement uncertainties and statistical sampling procedures. As soon as C/S measures become a part of the safeguards system, an overall performance figure or detection probability cannot be derived in such a manner. Therefore, the issue of how to assess the

performance of C/S equipment is being considered, since C/S devices have been applied in safeguards, in particular, as C/S equipment was and still is playing an ever increasing role in safeguards systems.

The paper results from an account of work performed for and in cooperation with the ESARDA Working Group on Containment and Surveillance.

## IAEA's Current Approach with Regard to Performance and Assurance of C/S Systems

After many years of discussion, still no suggestion can be made to assess the performance of C/S in quantitative terms comparable to those employed in the accountancy detection criteria, and there is a common understanding that there is very little likelihood of being able to develop such a method. The application of C/S involves human judgement in many respects and to a different extent, and, thus, is beyond any strict quantification.

The IAEA, in its safeguards implementation criteria, addresses the problem by categorising the types of C/S systems into single C/S systems and dual C/S systems. Single C/S systems consist of one or, to enhance system reliability, more devices of a functionally identical type. A dual C/S system consists of several C/S devices based on different physical principles, i.e., with no common failure mode.

The IAEA assigns different levels of assurance to the different types of C/S systems. As a general rule, nuclear material under C/S has to be re-measured to increase the assurance provided by the C/S system, even if the C/S systems are evaluated with positive results. The requirements for re-measurement are based on the type of C/S system and on the nuclear material category. The level of re-measurement required is lower for dual C/S systems, as they are attributed an added confidence compared to single C/S systems.

## Basic Concepts of the Proposed Approach

When considering possible ways to assess the suitability of C/S equipment, it becomes clear that performance and assurance are not device specific characteristics but have to be seen in the context of the whole safeguards approach in which this equipment is deployed. A perfectly efficient and reliable system will show bad performance for a given task when applied under the wrong conditions. For example, a powerful camera system monitoring the flow of casks does not show sufficient performance, if at the same time a statement on the content of the cask is needed, i.e., if the cask is empty, partially filled or full.
When C/S equipment is to be applied within a safeguards approach, first of all, a need has to be identified with which the C/S system should comply. The question then is, how to select the most appropriate equipment.

The approach chosen is based on a top-down procedure. In a first step, a task profile is developed. The underlying assumption is that the assessment of performance and assurance delivered by a system can only be carried out with regard to the expected function the system is intended to fulfil in the context of the whole safeguards approach. These expectations are outlined in the task profile. The task profile describes functional requirements to be met by C/S equipment without already having made a preliminary decision on the type of equipment.

In a second step, a performance profile of possible candidate C/S devices or combination of devices is established and matched against the requirements of the performance profile. Candidates that do not meet all of the mandatory requirements of the task profile can be immediately precluded from the further selection process. Many other task profile requirements can be met to a different extent, like for example the technical reliability of the devices, the effort and time needed for maintenance, service, and evaluation of results. Each device or combination of devices will generate a different performance profile that has to be compared to the task profile.

In a third step, the assurance profiles of different solutions can be compared and ranked to each other. Whereas the performance profile assures that all the data needed for safeguards purposes are generated, the assurance profile states if and to what degree these data may support the safeguards

conclusion. The IAEA already honours a combination of devices that generate the data based on different physical principles. Other aspects leading to differences in the assurance profile may be differences in the strength of data encryption or authentication, or the possibility of remote data transmission that may lead to increased unpredictability of data review by the IAEA compared to local storage and review of the same data.

## Establishment of the Task Profile

The task profile has to reflect, in detail, all requirements the C/S device has to fulfil within the safeguards approach. Hence, the information to be considered when establishing the task profile is similar to that required for establishing the safeguards approach:

- design information of the facility;
- facility operation;
- diversion and misuse assumptions;
- safeguards measures making up the safeguards approach.

As an exemplary facility, we have chosen a dry storage facility designed for storage of spent fuel casks as well as casks with vitrified highly radioactive waste (HAW) resulting from reprocessing of spent fuel assemblies. The storage capacity is designed for about 420 casks. HAW casks and spent fuel casks have similar designs. Empty casks may also be stored at the facility. The dimensions of the facility are about 200m by 40m with a height of 20m. The hall is divided into 2 parts, the cask reception area and the cask storage area.

The operational procedures foresee that casks are sealed in the shipping facility and received in the reception area of the storage facility. Here, casks are unloaded from the transport vehicle, prepared for storage and moved with a travelling crane into the storage area. At their storage position they are placed in an upright position onto the base plate. In case of a leakage, casks are moved back from the storage hall to the reception area for maintenance.

Diversion and misuse scenarios and, accordingly, the safeguards measures applied may vary with the situation in the state under consideration. If an Additional Protocol (AP) is in force and the State as a whole evaluated by the Agency with a positive conclusion, some scenarios may not be further regarded with the same relevance as in states without an AP in force.

Scenarios to be considered in the reception area during reception or maintenance work may consist of the following:

- removal of a cask after the receipt is recorded;
- declaration of a HAW cask as a spent fuel cask;
- replacement of a filled spent fuel cask with an empty cask, a dummy or a HAW filled cask.

Diversion scenarios in the storage area could be the following:

- to break through the outer wall and remove a cask;
- to break through the outer wall, remove a cask and replace it with an empty cask, a dummy or a HAW filled cask;
- to lift a cask, cut the bottom, and remove the content of the cask.

With regard to the safeguards approach, we can note that casks are regarded as items. Their nuclear material content was verified at the shipping facility, and this knowledge has to be maintained through appropriate C/S measures. Re-measurement of the cask content is not possible but only checking of identity and integrity. The system must be capable of distinguishing between spent fuel casks, HAW casks, and empty casks.

The safeguards approach provides for optical surveillance in the reception area to observe the flow of nuclear material in casks until the casks are transported into the storage area. Neutron detectors can be used to discriminate between loaded and empty casks. Seals are used in the storage area to identify casks and secure them against unauthorised removal.

To follow the movements of the casks in the reception area video cameras are installed at several positions in this area. The casks are under surveillance from entering the area until they are

transported into the storage area. Neutron detectors are used to record the movements of loaded casks. Detector recordings and surveillance recordings have to be correlated.

For storage, the casks are equipped at the reception with 2 seals (metal seals and electronic seals). The metal seal serves as a backup, whereas the electronic seal is used to check the identity and integrity of the casks during inspections.

## Exemplary Requirements Reflected in the Task Profile

For each step in the nuclear material (NM) handling procedures, the resulting requirements have to be considered and listed:

| | |
|---|---|
| NM handling procedure | Cask arrival in reception area<br>Prepare cask for unloading |
| Location | Reception area |
| Diversion assumption 1 | During transport the cask was diverted and replaced with a dummy cask<br>Normal declaration from receiver |
| Diversion indicator 1 | Cask identity not preserved |
| Diversion assumption 2 | During transport the cask was opened, NM content diverted and replaced with dummy material<br>Normal declaration from receiver |
| Diversion indicator 2 | Cask integrity not preserved |
| SG task profile (functional profile) | Allow to ascertain that cask identity and cask integrity were maintained during transport under following conditions:<br>- outdoor profile<br>- land bound transport profile<br>- operation profile<br>- general C/S device profile |
| Decision on C/S device class or combination of classes | Based on functional task profile a redundant sealing (metal seal / electronic seal) is foreseen. |

The exemplary task profile lists, for each step in the nuclear material handling procedures, all the requirements the C/S equipment has to cope with. For the transport of the casks from the shipping facility to the storage facility this could imply:

- Outdoor conditions
  - temperature range from –25°C to + 45°C
  - air humidity up to 100%
  - weather conditions: sun, dust, rain, fog, snow, hail
  - lighting conditions varying between day and night
- Transport conditions
  - rail and / or truck transport with according impacts (e.g., vibration, broken stones)
  - change of transport medium (reloading) must be possible, no interference with loading/reloading operations
- Operational conditions
  - failsafe (difficult to access cask content)
  - unattended operation during transport for at least x hours/days
  - ability to operate in SF cask vicinity (e.g., radiation level, decay heat)
  - no dependence on any external supply during operation (e.g., power, light, cooling)
  - in situ verification must be possible
  - probability of inconclusive outcome near zero
- General C/S device conditions
  - reliability

- detection probability
- false alarm probability
- probability of inconclusive outcome
- tamper resistance
- tamper indication
- data capture and storage
- evaluation effort (effort, skills, time)

In the first step, one should try to list the requirements as pure functional requirements and not already make a commitment for a special type of C/S equipment. In some situations, e.g., later in the storage area, the safeguards task could, in principle, be accomplished either by optical surveillance or by grouping casks together and applying a seal for a whole group of casks. The first step in establishing the task profile should not already predetermine the choice.

Based on the functional requirements, the class or the combination of classes of C/S devices capable of performing the required functions is identified, e.g., sealing systems, surveillance systems or radiation monitors, and the task profile requirements are then detailed with regard to the C/S device class chosen.

## Establishing the Performance Profile

The task profile determines the characteristics to be investigated for C/S equipment candidates under consideration. For each device or combination of devices the degree of compliance with the task requirements has to be established:

| Outdoor profile: | Requirement – Level | Performance Level Device 1 | Device 2 |
|---|---|---|---|
| Temperature range from –25°C to + 45°C | A | m | m |
| Air humidity up to 100% | H | m | m |
| Weather conditions: sun, dust, rain, fog, snow | H | m | m |
| Lighting conditions varying between day and night | A | m | m |
| Exposure time to outdoor conditions 10 weeks | A | m | n |

| Requirement Level |
|---|
| **A** mandatory |
| **H** high |
| **M** medium |
| **L** low |
| **N** Nice to have |

| Performance Level |
|---|
| **m** requirement met |
| **o** requirement almost met |
| **n** requirement not met |

In the example, we apply a graduation as shown for the strengths of the requirements and for the degree of fulfilment by the different devices. Mandatory requirements that are not met result in precluding the device concerned from further consideration. Questions that still have to be solved are how to rank alternative solutions that meet all mandatory requirements but show different degrees of fulfilment for graded requirements. A procedure to balance different requirements with each other has to be developed.

## Conclusions

A method to determine the performance of C/S equipment in quantitative terms with a sound mathematical approach is still not showing up, and there is a common understanding that there is very little likelihood of being able to develop such a method. The application of C/S measures involves human judgement in many respects and to a different extent and, thereby, is beyond any strict quantification.

In the proposed approach, a structured procedure is outlined that allows assessing the suitability of different C/S equipment to comply with the objectives of its application. The principle to determine the performance of C/S equipment will be to define a task profile and to check the performance profile against the task profile.

The method is under development and, in a first project phase and as an example, will be applied to a long-term dry storage facility of spent fuel assemblies to explore its capabilities in a practical application.

# Data Assurance of Operator-Owned Equipment for Safeguard Applications

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## Abstract:

*In the safe and secure operations of nuclear facilities, facility operators use a variety of sensors and sensor data. The data collected from these sensors is often the same data needed by safeguards inspectors but as the facility owned devices do not offer the level of authenticity required to draw safeguards conclusions. Inspectors must often duplicate these systems. If facility owned sensors included the required level of authentication, these sensors could be shared. This would lead to increased confidence of data authenticity of the facility owned systems and reduce installation costs when compared to installing a duplicate inspectorate owned system.*

*This paper will discuss the preliminary results of a study on the enhancement of data authenticity via an electronics platform for the secure transmission and recording of sensor data. This paper will also address the feasibility of the integration of secure sensor(s) into the operators' plant control systems such that inspectorates can branch operators systems while maintaining access to secured data streams.*

*The study includes the testing and evaluation of devices such as the Secure Sensor Platform (SSP) developed by Sandia in conjunction with safeguards equipment, and later expand to include operators' plant control systems. The SSP concept is a versatile, low-power/low-bandwidth solution for providing secure sensor collections that support multiple sensors. The SSP uses an electronics platform that is designed for monitoring a wide variety of, as well as multiple, sensors and for securely communicating the results to a host computer for analysis. The SSP will support standard algorithms for message authentication, initially as private key and potentially, in the near term, public key configurations. Additionally, the remote sensor platform incorporates active, passive and indicative tamper technologies, which are utilized to protect sensitive internal information.*

# Robotics for Advanced Storage Areas of Nuclear Materials: towards automated inventory and inspection of storage ponds

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***Abstract:***

*This paper presents the potentiality of robotics technologies for storage ponds of nuclear material through a demonstration prototype developed at the EC-Joint Research Centre (JRC) in the framework of JRC's R&D activities.*

*The presented robotics prototype, named SiLab Robotics Arm (SIROARM), is intended for the automated inspection of spent nuclear fuel canisters submerged in storage ponds using a reading system for sealing bolts developed at the JRC Seals and Identification Laboratory. This system is designed to be installed on a crane bridge and to replace the operator in the seals-checking process which is a difficult, tiresome and risky task.*

*Practicality of the implementation of this system in real-life environments is discussed including integration, maintenance, safety and operation aspects. Further enabling developments are also outlined. As a conclusion, the paper demonstrates that the implementation of robotics systems is feasible and that it brings advantages on operation and personnel safety but also in terms of storage security through permanent material control and limited access, which are a few relevant aspects for both plant operators and safeguards authorities.*

**Keywords:** robotics; inspection; inventory; storage ponds; automation.

## 1. Project Background

The SiLab Robotics Arm (SIROARM) project concerns the development of a robotic manipulator arm prototype (see Figure 1) to carry out automated reading of sealing bolts on spent nuclear fuel canisters submerged in a storage pond. The aim of this project is to study the technical feasibility and demonstrate the potential of automating inspection tasks on ultrasonic sealing bolts.

This project was carried out in the framework of the JRC Nuclear Safeguards R&D program and is the result of a fruitful collaboration between two units of the JRC Institute for the Protection and Security of the Citizen (IPSC): the Traceability and Vulnerability Assessment (TRVA) unit and the Nuclear Safeguards (NUSAF) Unit.

The project started in year 2003 and was completed in 2005 with several demonstrations of the first prototype to Euratom DG-TREN and IAEA representatives.

Figure 1: SIROARM and control unit

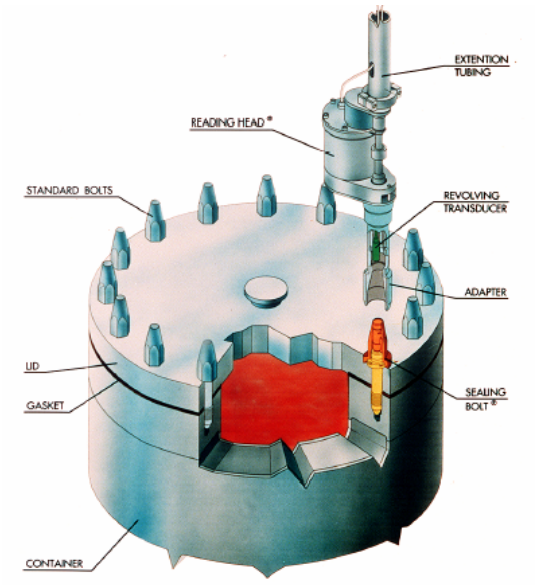


Figure 2: SiLab sealing bolt and reading head system on a nuclear container

The JRC technology developed at the SiLab group of the IPSC-TRVA unit and used on sealing bolts is based on ultrasonic signature reading and identification. Figure 2 shows a nuclear container, standard sealing bolts on its cover, and a reading head. The sealing bolt is read underwater and is suitable for applications inside reactors or in aggressive environments. The machined top surface of a sealing bolt for a cover of a nuclear container is the relevant part of the ultrasonic seal. After its installation on a container, this bolt surface has a specific ultra-sound signature which is recorded by the plant operator. This signature is irremediably altered if the bolt is removed.

The seal-reading system developed at SiLab comprises a motorized reading head, a PC laptop data acquisition system and a recording and identification software. The ultra-sound signature is recorded after-installation and used as identification pattern for further signature readings. The seal-reading head has a weight of 5 kg and is connected to a 7-8 meters-long stick that weights approximately 10 kg when partially submerged in the storage pond.

These ultrasonic based sealing bolts are installed in the La Hague (FR) and Sellafield (UK) nuclear facilities and are routinely inspected. Figure 3 shows the Sellafield storage pond where hundreds of these types of bolt-seals are installed.

Periodically, nuclear safeguards inspectors from international (i.e. IAEA, Euratom) or national authorities, have to carry out inspection activities consisting of verifying operators declarations in terms of nuclear material. A typical task consists of checking the integrity of seals placed on containers for nuclear material.

The operator usually stands on a bridge crane which is approximately positioned above the container to inspect with a +/-0.6m of accuracy. Once positioned, the operator has to visually place the seal-reader on the sealing bolt from a height of 4-5 meters, and exert enough force to adjust the reader to the bolt. This operation is potentially unsafe and uncomfortable since the operator has to tilt his body on the crane banister, handling the stick (with the reader attached) that weights approximately 15 kg and that is partially submerged in the pond.

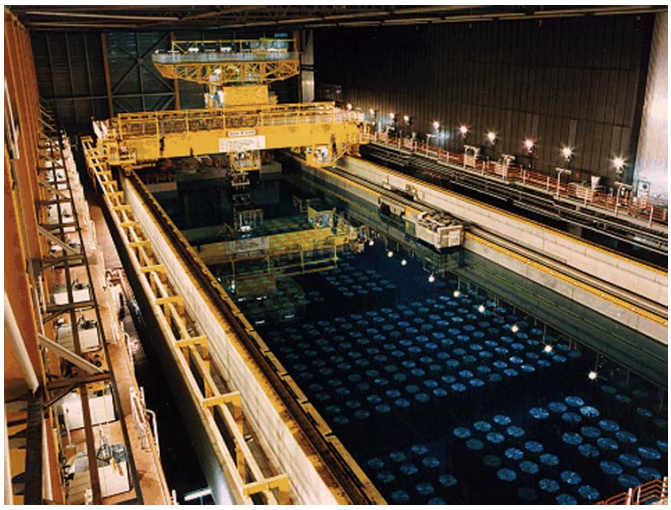

Figure 3: Sellafield storage pond

As a consequence from the working conditions and from the sharp increase of the quantity of stored material in the following years, the use of a robotics system for inspecting sealing bolts would be justified in terms of the reduction of workers hazards and risks, of manpower costs, and of improvement of inspection coverage since the robotics system could in principle carry out automated and frequent inspections cycles. In addition, the results quality of the ultrasonic bolt-seal reading process would benefit from a precise and tuned insertion of the reading head by an automated system.

## 2. Main parts of the SIROARM robotic system

The SIROARM manipulator is a SCARA-type of manipulator with 3 degrees of freedom (DOF) for positioning the seal reading head on sealing bolts. These 3 DOFs are a vertical translation motion, a shoulder rotation about a vertical axis, and an elbow rotation about a vertical axis. The shoulder and the elbow move two substantially horizontal links. Figure 4 shows an outline and a drawing of the manipulator arm. Cinematically, there is no need of orientation degree of freedom since:

- the seal reading head attached to the stick is symmetrical with respect to its main axis which is theoretically perpendicular to and intersecting the last arm link;
- although the stick can be slightly bent, an elastic joint can be placed in between the stick and the reading head to accommodate for misalignments during the ‘reading-head to sealing bolt’ insertion phase;
- the reading head is internally conical and therefore independent from the bolt head rotation.

The SIROARM system comprises:

- the manipulator arm to which a stick is attached;
- at the lower end of the stick, three sensory devices are attached:
    - a camera coupled to an artificial vision system to identify the sealing bolt and to guide the robot approach to it
    - a force-torque sensor on the arm end-effector to sense reaction forces and torques during the insertion phase;
    - a reading head for sealing bolts attached to a stick;
- a mobile power and control cabinet with dual battery and grid-based power supply; the cabinet includes the motion control computer, the vision system computer and is prepared to

accommodate and connect the PC-laptop of the inspector for the data acquisition and identification;

- a touch-screen user interface to control the manipulator, the reading head and the identification process.

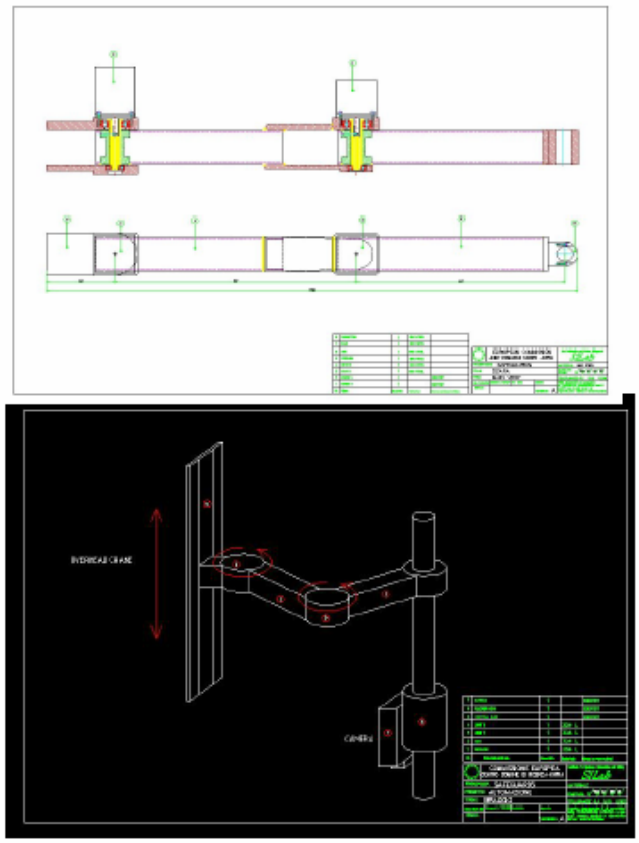

Figure 4: 2D and 3D preliminary drawing of the manipulator

As there were not available any suitable commercial robotic system for this application and in order to obtain a fast and low cost prototype, the following decisions were taken:

- to design and manufacture the mechanical structure of the manipulator arm at SiLab taking into account the requirements of rigidity, robustness and light weight of the mechanical structure; the SiLab group has a mechanical workshop with numerically controlled machines and CAD design stations;

- to base the software control system on the JRC GENERIS software, a generic motion control platform for robotics, developed at the Rialto Laboratory of the JRC NUSAF unit and used in many robotics systems including mobile platforms, industrial robots, teleoperated articulated manipulators, and Cartesian Gantry manipulators;

- to use the VEA industrial vision system and JR3 force/torque sensor that were already interfaced with GENERIS in past projects;

- to select latest industrial Components-Off-The-Shelf (COTS) for motors, gears, computers and other electronics/electrical/electro-mechanical components;

- to avoid software interface developments between the GENERIS HMI, the vision system and the seal-reading system;

- to avoid the development of water-proof enclosures for the camera to be placed near and parallel to the reading head and other arm components;

- to avoid sensorial redundancy usually implemented for safety reasons.

## 3. Main characteristics

### 3.1. Joints travel and speed capabilities

Table 1 gives the travel ranges of every joint and their speed capability.

| Joint | Travel range | Max Speed |
|---|---|---|
| Arm elevation | 0-1100 mm | 28 mm/sec |
| Shoulder | [-100; +100] deg | 30 deg/sec |
| Elbow | [-100; +100] deg | 30 deg/sec |

Table 1

### 3.2. Payload

The arm is designed to have payload capacity of 20 kg, comprising the stick and reader weight plus a bolt-reader insertion/extraction force of 5 kg.

### 3.3. Workspace

The workspace is given by the links length and joints range, and has a semi-cylindrical shape of 900mm (radius) x 1100mm (height).

### 3.4. Accuracy

The reading head can be place above the bolt with an accuracy of +/- 5 mm. The arm is designed to be accurate better than +/-1mm in its entire workspace. However, the robot installation has to be done properly in such a way that the verticality error be less than 0.3 degrees.

### 3.5. Arm Structure

As in nuclear environments stainless steel is the required material for mechanical structures in contact with water, the horizontal links of the arm were are made of it using a box-type section shape. However, the vertical link was designed using a standard industrial component mainly made of Aluminium, in which a linear guide driven by with a ball-screw axis is mounted. The whole arm weight is about 50kg.

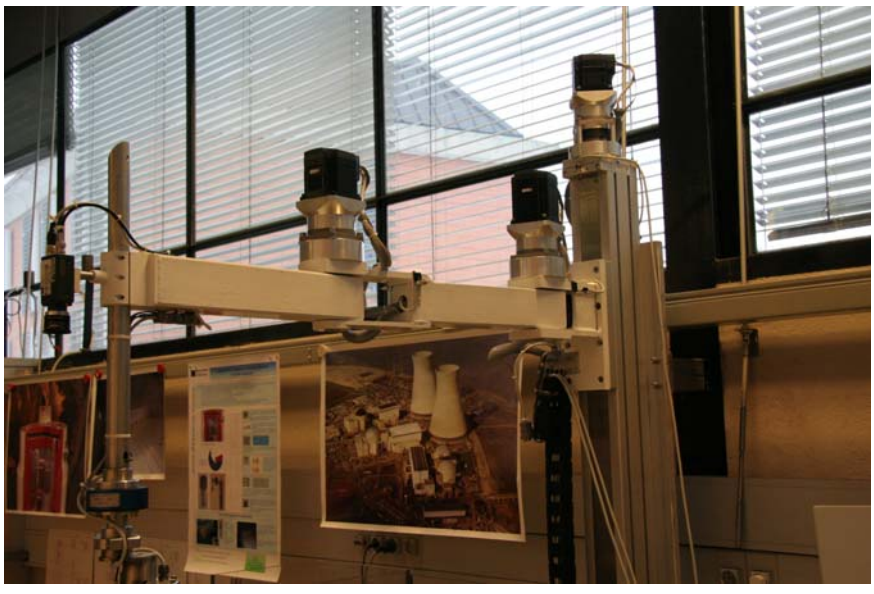

Figure 5: SIROARM manipulator joints

## 4. The vision system

The HVQ industrial vision system supplied by VEA (www.vea.it ) was used for guiding the robot to the sealing-bolt. A monochrome camera was used and mounted on the manipulator tool flange. The camera has a C-mounting lens that has to be adjusted to focus the object of interest from a given distance.

Figure 6: Camera mounted on the tool flange

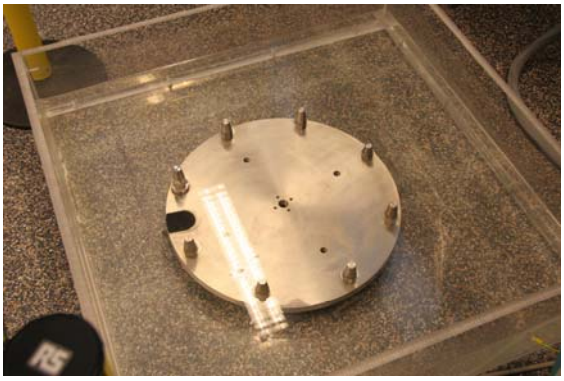

Figure 7: Container cover and recess fixture

The VEA identifies pre-taught patterns and gives their position and rotation with respect to the image reference frame. This information is fed through a serial line to the robot controller and guides the robot during the approach phase.

A number of considerations have to be taken into account in the implementation of a vision system for this application:

- The image processing is very sensitive to light conditions and this is one of the primarily reasons of object detection failure. In fact, the reference pattern of an object is made of a number of pixels of different intensity with a specific distribution; if the lighting conditions change, the intensity distribution of the same object become different and the pattern cannot be matched. In our case, as the surfaces of the container cover and of the bolts are very reflective, the sealing bolt cannot be a proper pattern to identify. Therefore, the solution is to use a fixture which is independent from lighting conditions, as for instance, a dark recess on the container cover (see Figure 7). As the sealing bolt is always placed on one side of this recess, its position can be accurately defined. An additional fixture could be a coloured ring on the sealing bolt.

- Since the lens has a fixed regulation, the image remains focused in a few cm range; therefore, the sealing bolt localization can be done only from a given height and preferably from a distance of 0.5 to 1.5 meters to limit the detection error to less than 1mm. In our experimental set-up, the vision system takes a picture from 1.5 m and the robot approach is accurate enough.

- In the final application, the camera shall be submerged since the container top is usually at 3-4 meters of depth from the water surface. The camera shall be calibrated in similar conditions to compensate for the underwater image distortion.

- A radiation-tolerant camera shall be used in the final application.

## 5. The force-torque sensor

A 6 DOF force-torque sensor from JR3 Inc was installed between the reading head and the stick. This force-torque sensor is used for:

- horizontal adjustments during the insertion of the reading head on the sealing bolt

- controlling the pushing force on the sealing bolt
- stopping motion in case of detection of abnormal forces.

In the final system, a waterproof version of this sensor shall be used with external electronics to be radiation tolerant.

Figure 8: JR3 6DOF force-torque sensor

Figure 9: Power and control cabinet

## 6. Installation of the robotic system

In order to minimize the installation costs, the SIROARM system is intended to be mounted on the trolley of a horizontal linear guide installed along a bridge crane. The trolley shall be movable by hand and shall be lockable with a manual clamp.

The robot base, consisting of its vertical axis, needs strong and stiff fixation means on a vertical support connected to the linear guide in order to ensure its stability. The verticality error of the first axis shall be better than 0.3degrees.

In this configuration, the operator shall command the bridge crane and move by hand the manipulator approximately above the container to be inspected. This rough positioning must be done with an accuracy of +/-1m to enable the automated reading by the manipulator.

In a fully automated system, the linear guide shall be motorized and equipped with an absolute linear position sensor. The bridge should also have an absolute positioning system. The horizontal linear axis, the bridge crane and the manipulator shall be all controlled from a single fixed cabined and motion control system. Therefore, this advanced configuration would require a 2D mapping of the pond giving the coordinates of every container. However, the implementation of such a fully automated solution would require significant and expensive modifications of an existing bridge crane installation.

## 7. Conclusions

The design and development of this first manipulator prototype was a fruitful experience demonstrating the technical feasibility of automated bolt-seal reading in a laboratory simulated environment. This experience also allowed to establish requirements for its implementation in a nuclear storage pond, that would be advantageous for operation and personnel safety but also in terms of storage security through permanent material control and limited access.

# Session 12

# NDA-II gamma spectrometry

# Uranium enrichment assay with a $LaBr_3$(Ce) scintillation detector: A promising option for the 2nd generation of COMPUCEA

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***Abstract:***

*A careful evaluation of uranium enrichment measurements with a 2" x 2" $LaBr_3$(Ce) scintillation detector has been carried out in order to test this new type of detector as a room-temperature detector option for future use in the in-field analyses made with COMPUCEA. Spectrum analysis has been made with an adapted version of the NaIGEM analysis code. Comparative performance data obtained from measurements with HPGe, NaI(Tl) and $LaBr_3$(Ce) detectors on low-enriched uranium reference samples are presented.*

**Keywords:** uranium enrichment; gamma spectrometry; lanthanum bromide detector; NaIGEM analysis code

## 1. Introduction

The acronym COMPUCEA (COMbined Procedure for Uranium Concentration and Enrichment Assay) refers to a measurement procedure for the determination of the uranium elemental concentration and $^{235}$U isotopic abundance ("enrichment") in relatively small samples of unirradiated uranium product materials. The applied measurement techniques include absorption edge spectrometry for the uranium concentration analysis, and passive high-resolution gamma spectrometry for the $^{235}$U enrichment determination. The respective measurements are made on site in European Low-Enriched Uranium (LEU) fuel fabrication plants by analysts from the Institute for Transuranium elements (ITU), Karlsruhe, in support of joint Euratom/IAEA Safeguards inspections during the physical inventory verification (PIV). This measurement support from ITU with COMPUCEA during the PIV campaigns has been provided for more than a decade with the 1$^{st}$ generation of equipment, consisting of a radioisotope-based K-edge densitometer and a HPGe-based high-resolution gamma spectrometer [1, 2, 3].

Recently, ITU started to develop a so-called 2$^{nd}$ generation of equipment for COMPUCEA. One of the objectives for the redesign aimed at replacing the liquid nitrogen-cooled HPGe detectors by room-temperature detectors in order to simplify the equipment for the in-field measurements. This goal has been fully achieved, without loss in performance, for the uranium concentration measurement part of COMPUCEA, where the change from K-edge densitometry to L-edge densitometry allowed to replace the previous HPGe detector by an ultra high-resolution Si detector operated under modest Peltier cooling [4, 5]. As for the enrichment measurement part of COMPUCEA, searches for an adequate substitute for the high-resolution HPGe detector are still under way.

With the recently discovered Ce-doped lanthanum halide ($LaCl_3$, $LaBr_3$) scintillators [6], a new type of scintillation detector is now becoming available, which is opening up good prospects for high-quality enrichment measurements with a room temperature detector. In this paper, we report on a performance study for enrichment measurements with a $LaBr_3$(Ce) scintillation detector, addressing the specific application to small-size uranium sample measurements as encountered in the in-field analyses made with COMPUCEA.

## 2. Enrichment assay in COMPUCEA

Before describing the application of the new $LaBr_3(Ce)$ scintillation detector as a potential detector alternative, it might be helpful to briefly recapitulate the specific measurement conditions and procedures for the enrichment measurements with COMPUCEA. Three measurement conditions ultimately determine the currently applied method for the gamma-spectrometric enrichment measurement:

1. The physical inventory samples provided by the inspectors are of small quantity, typically a single uranium pellet or a few grams of oxide powder. This automatically rules out the classical type of enrichment measurement made in the "infinite thickness" geometry.
2. Since the measurements serve the purpose of accountancy verification, they are calling for the highest possible measurement accuracy with small to negligible systematic measurement errors. It seemed questionable whether this requirement could be fully met by the alternative measurement option for enrichment measurements based on the intrinsic assay approach [7].
3. The parallel measurements with COMPUCEA for the determination of the uranium element content made by L-absorption edge spectrometry requires a dissolution of the solid uranium samples, because nondestructive measurement techniques allowing a direct determination of the uranium concentration from a solid uranium sample with high accuracy unfortunately do not exist up to date.

With the uranium samples prepared in liquid form, the enrichment measurement with COMPUCEA follows the procedure of counting the most prominent $^{235}U$ gamma ray at 186 keV from a well-defined volume of sample solution in a well-defined and calibrated counting geometry as shown in Fig. 1. The gamma counting is performed in a well detector to achieve the highest possible detection efficiency. This approach allows accurate enrichment measurements, provided (i) the sample volume is kept constant, (ii) the uranium content in the measurement sample is accurately known, (iii) the matrix of the measurement solution is well controlled, and (iv) the counting set up is calibrated against representative reference solutions.

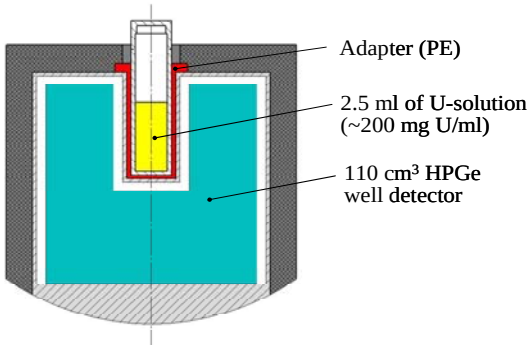


*Fig. 1: Present counting configuration for the $^{235}U$ enrichment measurement in COMPUCEA with a HPGe well detector.*

An exact volume of 2.5 ml of uranium solution, precisely determined with an accuracy of 0.02% from a combined sample weighing and density measurement, is transferred into a cylindrical plastic vial with an inner diameter of 11 mm for counting in the well detector. The uranium concentration of the solution, and hence the amount of uranium in the measurement sample, is known from the parallel densitometry measurement of COMPUCEA with an accuracy better than 0.2%. The uranium concentration and the nitric acidity of the solution matrix are established within narrow limits - through strict procedures followed during sample dissolution and subsequent dilution – at nearly constant values of 200 mgU/ml and 3M $HNO_3$, respectively. Small corrections for gamma self attenuation are applied for samples slightly deviating from these nominal values. These corrections also take into account the additional self attenuation effects caused by gadolinium, if present. Information on the Gd content will in the future be obtained from the L-edge densitometer operated in the XRF mode [4].

The type of detector used up to now during the in-field measurements is a 110 $cm^3$ HPGe well detector with a 16 mm diameter x 40 mm deep well in the detector cap. For 3 different units of this type of detector, the detection efficiency for 186 keV gammas in the counting configuration shown in Fig. 1 was determined to 26.4, 28.3 and 31.6%. This yields count rates between 57 and 68 cps per %

enrichment for the typical amount of 0.5 g of uranium contained in the 2.5 ml of sample solution. This rate necessitates a counting time of about 20-25 minutes to reach a measurement precision of 0.2% for a sample enrichment of 3%.

For highest measurement accuracy and reliability, enrichment measurements made with the high-resolving HPGe detector undoubtedly offer the best choice. Practical drawbacks for the in-field measurements, however, are the need for liquid nitrogen for detector cooling, and the relatively long cooling times for the well-type detector. For the given HPGe well detector it takes about 24 hours before the detector has sufficiently cooled down to reach the nominal energy resolution. During the recent re-design of the COMPUCEA equipment towards a more compact and ready-to-use type of instrumentation we have therefore also evaluated detector alternatives, which do not require cooling.

The use of CdZnTe solid state detectors has been ruled out a priori because of their small volumes and corresponding low detection efficiency. Realistically, this leaves only the use of a suitably dimensioned scintillation detector as a viable detector option. In a first approach we have carried out a performance assessment from in-field measurements with a 3" x 3" NaI(Tl) well detector, which with its 16 mm well diameter offered the same counting geometry as the HPGe well detector. Despite the much poorer energy resolution of the NaI detector, the results obtained showed acceptable performance [5].

It is a fortunate situation that, with the recent discovery of the new lanthanum halide scintillators, the gamma spectroscopy community is now provided with a promising new type of room temperature scintillation detector with distinctly improved performance compared to the well-known NaI(Tl) detector.

## 3. Features of $LaBr_3$(Ce) scintillation detectors

The new cerium-doped lanthanum chloride, $LaCl_3$(Ce), and lanthanum bromide, $LaBr_3$(Ce), scintillation detectors, manufactured by Saint-Gobain Ceramics & Plastics Inc. and now commercially available under the registered trademarks BrilLanCe®350 and BrilLanCe®380, respectively [8], have the potential of significantly enhancing the performance and capabilities of scintillator-based gamma spectroscopic systems in many areas. The application to $^{235}U$ enrichment measurements for Safeguards purposes as discussed in this paper represents just one example.

The production of larger-sized lanthanum halide crystals has rapidly progressed during the last 2-3 years. Standard detector sizes up to 3" x 3" are now available, although detector prices are still relatively high (~200 € /$cm^3$). Existing performance data show about 20-30% better energy resolution for the lanthanum bromide than for the lanthanum chloride detectors. We therefore chose the former one for our studies. Some of the noteworthy practical advantages of the new $LaBr_3$(Ce) scintillator, compared to the traditional NaI(Tl) scintillation detector, are:

- Improved energy resolution;
- About 10 times faster light output decay, enabling high count rate applications;
- High temperature stability;
- Higher gamma stopping power and detection efficiency.

| Property | $LaBr_3$(Ce) | NaI(Tl) |
|---|---|---|
| Crystal density ($g/cm^3$) | 5.29 | 3.67 |
| Thickness for 50% attenuation of 662 keV gammas (cm) | 1.8 | 2.5 |
| Light yield (Photons/keV) | 63 | 38 |
| 1/e decay time (ns) | 16 | 250 |
| Temp. coefficient of light output (%/°C between 0° and 50°C) | <0.02 | -0.3 |

**Table 1**. Comparison of scintillator detector properties [from [8]).

Table 1 lists comparative values for some pertinent physical properties of $LaBr_3(Ce)$ and NaI(Tl) detectors. Comparative data for the energy resolution are given in Table 2. The quoted energy resolutions were measured for the actual 2" x 2" $LaBr_3(Ce)$ detector used in the present study, and for a NaI(Tl) detector of same dimensions with a nominal energy resolution of 6.8% at 662 keV. The right-hand column in Table 2 gives the improvement factor for the energy resolution offered by the $LaBr_3(Ce)$ detector. Better energy resolution for the lanthanum bromide detector is obtained over the whole energy range covered, with the improvement factor increasing with increasing gamma energy. For the 186 keV gamma ray, the lanthanum bromide detector offers two times better energy resolution than the NaI detector.

| Energy (keV) | FWHM energy resolution (%) | | Improvement factor for $LaBr_3(Ce)$ |
|---|---|---|---|
| | $LaBr_3(Ce)$ | NaI(Tl | |
| 29 | 13.58 | 20.10 | 1.48 |
| 59 | 9.80 | 14.65 | 1.50 |
| 88 | 7.55 | 11.88 | 1.57 |
| 186 | 4.87 | 9.71 | 1.99 |
| 662 | 2.68 | 6.50 | 2.43 |
| 1001 | 2.09 | 5.46 | 2.61 |
| 2614 | 1.34 | 3.55 | 2.65 |

**Table 2.** Measured energy resolutions for 2" x 2" $LaBr_3(Ce)$ and NaI(Tl) detectors.

The spectral appearance of gamma spectra from low-enriched uranium is displayed in Fig. 2. The figure shows spectra for a typical COMPUCEA sample (0.5 g of uranium with an enrichment of 4.4% in 2.5 ml of nitric acid) taken with 3 types of detectors: the 110 $cm^3$ HPGe well detector so far routinely used in COMPUCEA, a 3"x3" Na(Tl) well detector, and a 2"x2" $LaBr_3(Ce)$ detector. The measurement example produced with the lanthanum bromide scintillation detector shows a uranium gamma spectrum, in which the main $^{235}U$ gamma rays (143, 163, 186, 205 keV) are reasonably resolved.

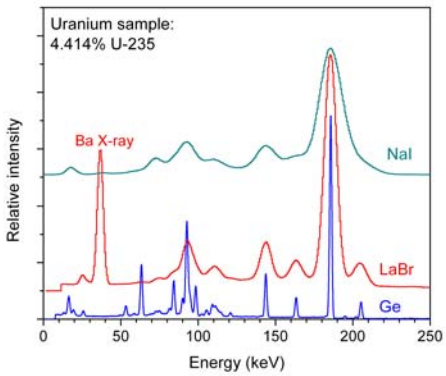


*Fig. 2. Comparison of HPGe, $LaBr_3(Ce)$ and NaI(Tl) gamma spectra from a low-enriched uranium sample.*

A peculiarity of the lanthanum-based scintillators is the presence of an inherent background originating from the electron capture and β-decay of the long-lived minor La isotope $^{138}La$ ($t_{1/2}$ = 1.05x10$^{11}$y) to $^{138}Ba$ (66.4%) and $^{138}Ce$ (33.6%, see schematic decay scheme in Fig. 3). The Ba X-ray peak visible in the $LaBr_3$ spectra in Figs. 2 and 3 results from the EC decay of $^{138}La$ to $^{138}Ba$ (the single X-ray peak near 37.4 keV in fact represents the accumulated energy given off by the X-rays in the EC process). The β-decay to $^{138}Ce$, in turn, manifests itself in a beta continuum background with an endpoint energy of 255 keV as shown in Fig. 3. This continuous beta background occurring in the energy range of the $^{235}U$ gamma rays normally turns out to be negligible for enrichment measurements on bulk uranium samples, but it may notably contribute (about 20-30%) to the Compton and Bremsstrahlung background from the high-energy $^{234m}Pa$ gamma rays in enrichment measurements on small quantities of uranium (1 g or less as in the case of the COMPUCEA measurements). This is illustrated in Fig. 3, which shows the magnitude of the internal beta continuum background in relation to the

gamma spectrum produced by a 5 ml uranium solution sample with 1 g of natural uranium (kept at 1 cm distance from the surface of a 2" x 2" $LaBr_3$ detector). Fortunately, even in the case of the low-activity COMPUCEA samples, the added detector background component in the energy range up to 250 keV is properly accounted for in the spectrum evaluation and has proved to have a negligible effect on the performance of the enrichment assay (Note: An elevated detector background due to radiations from the decay of $^{138}La$ also occurs in higher energy regions, outside of the region of interest for enrichment measurements).

It should be noted that for the efficient detection of 186 keV gamma rays lanthanum bromide crystals do not need to be thicker than about 2-2.5 cm. A 2 cm thick $LaBr_3$ crystal doped with 5 wt% Ce stops 95% of 186 keV photons. A standard crystal thickness of 1" is therefore more than adequate for enrichment measurements. The reduced detector volume would also reduce the internal detector background.

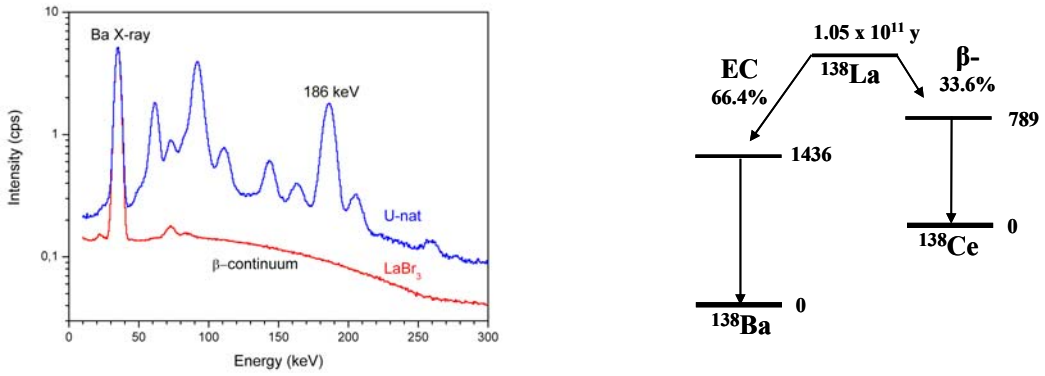


*Fig. 3. Magnitude of internal background from a 2" x 2" $LaBr_3$ detector in comparison to the gamma spectrum from a 1 g U-nat sample.*

## 4. Application to enrichment measurements on COMPUCEA samples

Sample counting in a well counter as routinely practiced so far in the COMPUCEA measurements with the HPGe well detector not only offers high detection efficiency, it also represents a counting configuration being least sensitive to measurement variabilities such as sample positioning or small dimensional variations of the sample containers holding the uranium solutions. Since lanthanum bromide detectors are not yet available in the form of well detectors, a different counting geometry as shown in Fig. 4 had to be chosen for the test measurements with the $LaBr_3$(Ce) detector.

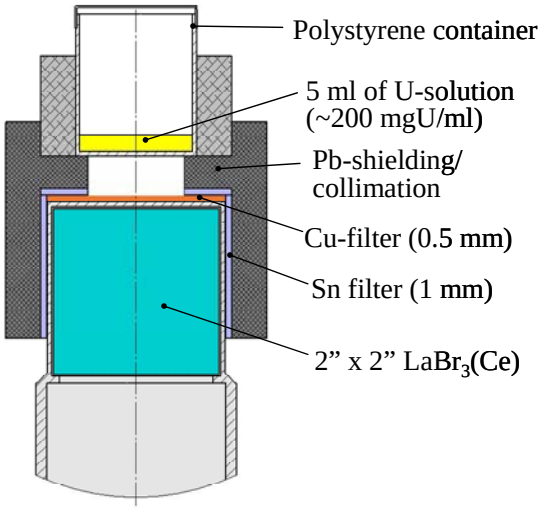


*Fig. 4. Counting configuration used for the measurements with the $LaBr_3$ detector.*

The sample-detector configuration represents a compromise between closest possible sample-to-detector distance for maximum detection efficiency, and desirable larger sample distance for increased immunity to variations in sample and container dimensions. To partly compensate for the somewhat lower detection efficiency, the volume of the uranium solution was increased from 2.5 ml to 5 ml. For the polystyrene sample container with an inner diameter of 36 mm this yields a 5 mm thick solution layer. The sample was viewed by the detector through a 10 mm high, 30 mm diameter Pb collimator and a 0.5 mm thick Cu filter.

Table 3 compares the count rates for the 186 keV gammas obtained with the configuration shown in Fig. 4 with the corresponding rates measured in the standard 110 $cm^3$ Ge well detector of COMPUCEA, and with an alternatively tested 3" x 3" NaI well detector. It is obvious that the less efficient counting geometry of Fig. 4 provides lower peak count rates than the well-type configurations. For the same amount of uranium the counting efficiency decreases by about a factor of 4. A counting configuration with significantly enhanced detection efficiency, however, could be realized with available $LaBr_3$ detectors in the form of a sandwich-type configuration, incorporating, for example, two 2" x 0.75" $LaBr_3$ detectors as sketched in Fig. 5. Such a configuration would provide a relatively robust and efficient counting geometry, yielding 186 keV count rates of about 150 cps per % enrichment from a 5 ml sample as indicated in the right-hand column of Table 3. Graphs showing the expected measurement precision from this 186 keV count rate in dependence on counting time and enrichment are given in Fig. 6.

| Detector/ Configuration | 110 $cm^3$ HPGe well (Fig. 1) | 3" x 3" NaI well (Fig. 1) | 2" x 2" $LaBr_3$ planar (Fig. 4) | 2" x 0.75" $LaBr_3$ sandwich (Fig. 5) |
|---|---|---|---|---|
| Sample vol. (ml) | 2.5 | 2.5 | 5 | 5 |
| Amount U (g) | 0.5 | 0.5 | 1 | 1 |
| Filter | 1 mm PE | 1 mm Ti | 0.5 mm Cu | 0.5 mm Cu |
| 186 keV (cps) per % enrichment | 60, 65, 72 (rates from 3 units) | 109 | 36 | ~ 150 (expected) |

**Table 3**. Comparison of 186 keV peak count rates from different measurement configurations.

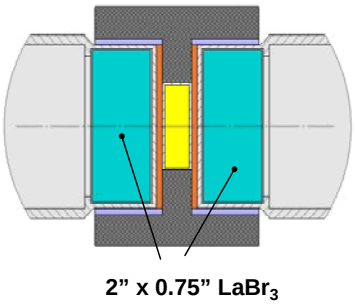


*Fig. 5. Proposed sandwich detector configuration for enhanced detection efficiency. Sample volume 5 ml with 1g of U.*

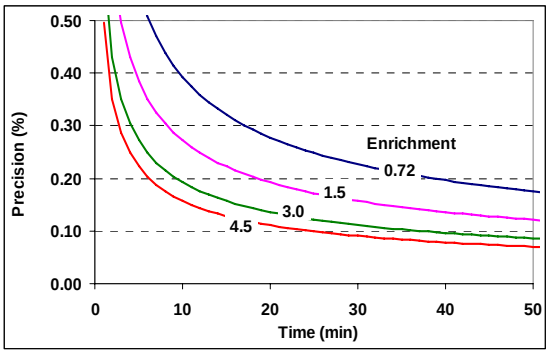


*Fig. 6. Expected counting precision for the configuration of Fig.5.*

## 5. Spectrum analysis

The method for spectrum analysis applied to the $LaBr_3$ spectra was adopted from that used in the NaIGEM code for $^{235}U$ enrichment measurements with NaI detectors [7]. In this method, response profiles are computed for each of the components contributing to the 130 - 290 keV region of the spectrum that is analyzed. These responses are then fit to the observed data by the method of least-squares. Two of the components, of course, are spectrum profiles for $^{235}U$ and $^{238}U$. However, other profiles are generated to account for small shifts in the spectrum gain and changes in the detector resolution that may have occurred.

The peak structure in the 130 - 290 keV region is primarily due to $^{235}U$ gamma rays. The principal gamma ray peaks of $^{235}U$ are: 143.76, 163.37, 182.57, 185.72, 194.94, 202.1, and 205.33 keV. Although gamma rays from $^{238}U$ and its daughters are very weak in this region, their Compton scattering and Bremsstrahlung radiations do contribute significantly to the background continuum. In the case of the $LaBr_3$ detector the internal background produced by the detector material itself contributes to the background.

The principal shape of the gamma ray peaks for both the $LaBr_3$ and NaI detectors is described by Gaussian functions. The $^{235}U$ response profile is generated using known gamma-ray branching intensities that have been modified to account for attenuation by filters and other materials and by the relative detection efficiency. The equation describing the "background" continuum contains two components. The first accounts for the "step" function associated with the different background levels in front of and after the peaks. The other is a polynomial equation used to describe the shape of the Compton plus Bremsstrahlung continuum associated with the decay of $^{238}U$ and its daughters, and with the internal background continuum of the $LaBr_3$ detector. The method of analysis is iterative. Convergence must first be reached to account for changes in gain and resolution before a final calculation is made to determine the intensity of the $^{235}U$ response. An example showing the final fit of the principal $^{235}U$ response from the $LaBr_3$ detector is given in Fig. 7. The fitting graph in Fig. 8 shows another fitting option in which the 186 keV peak is isolated from the rest of the $^{235}U$ response. The $LaBr_3$ spectral responses displayed in the figures, if compared with the corresponding response from a NaI detector shown for comparison in Fig. 9, best illustrate the kind of spectral improvement gained with a lanthanum bromide detector.

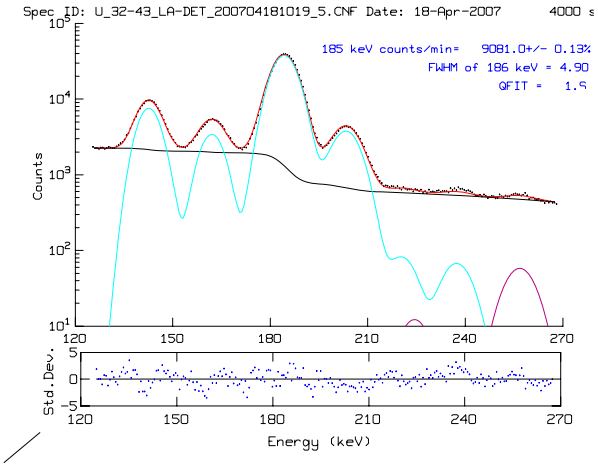


*Fig.7. $LaBr_3$ graph fitting the principal $^{235}U$ response.*

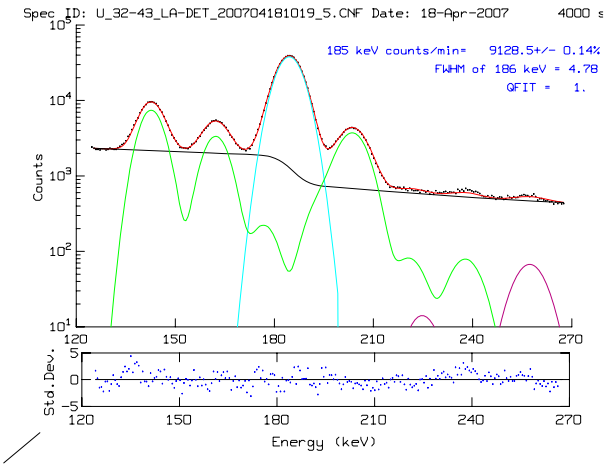


*Fig.8. Another fitting version for $LaBr_3$ spectra where the 186 peak is isolated from the rest of the $^{235}U$ response.*

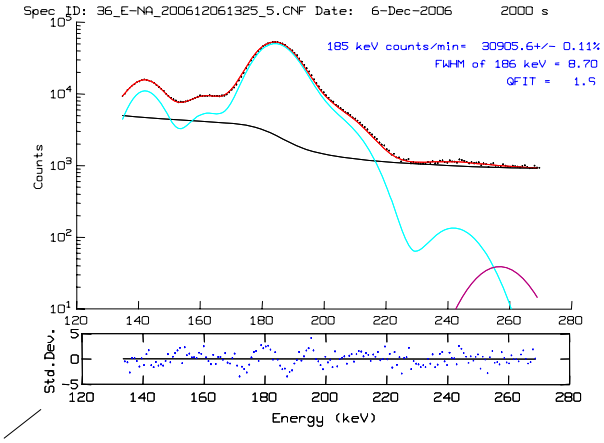


*Fig 9. Fitting graph for a NaI spectrum.*

## 6. Performance

In a perfect enrichment measurement, the registered 186 keV peak count rates are expected to be strictly proportional to the $^{235}U$ isotopic abundance. This feature, if realized, not only simplifies calibration, but also helps to reduce and/or to avoid systematic measurement errors. Achieved strict proportionality primarily proves the ability of the applied method for spectrum analysis to extract unbiased net peak counts from the measured gamma spectra, irrespective of given spectral features like, for example, the degree of energy resolution or changing peak-to-background ratios.

In order to verify the assumed proportionality, we have carried out high-precision measurements on a set of carefully characterised uranium reference solutions with 6 different grades of $^{235}U$ enrichment (0.3158, 0.7202, 1.3802, 2.1039, 3.2469 and 4.3138 wt% $^{235}U$). Comparative measurements were made with 3 different units of the standard COMPUCEA HPGe well detector, with a 3" x 3" NaI well detector, and with the 2" x 2" $LaBr_3$ detector. From each of the 6 reference solutions 2.5 ml aliquots were transferred into the cylindrical sample vials used for the counting in the well detectors (Ge and NaI), and 5 ml aliquots were loaded into the polystyrene sample containers for the counting with the $LaBr_3$ detector in the configuration shown in Fig. 4. The combined uncertainties for the uranium concentration (from gravimetry), for the $^{235}U$ isotope abundance (from mass spectrometry), and from sample weighing yielded a total uncertainty of 0.12% (1s) for the amount of $^{235}U$ contained in the samples with enriched uranium. For the samples with depleted (0.3158%) and natural uranium the accuracy of the $^{235}U$ content was limited to 0.30 and 0.15%, respectively, by the respective uncertainties of the mass spectrometry measurements for the $^{235}U$ abundance.

For all gamma measurements, the statistical counting error for the 186 keV gammas was kept below 0.1%. Since the reference solutions did not have exactly the same uranium concentration (ranging between 193 and 201 mgU/ml), small corrections for gamma self attenuation had to be applied to the measured 186 keV count rates. The differential change of the count rate as a function of uranium concentration around a reference concentration of 200 mgU/ml has been determined from measurements made on samples of same enrichment but different uranium concentration. The observed differential changes are listed in the 1$^{st}$ row of Table 4. Another correction made to the measured 186 keV peak rates accounted for small contributions of gamma rays from $^{234}Pa$ and $^{234m}Pa$ to the 186 keV line from $^{235}U$. The respective corrections amount to 0.38% and 0.16% for the depleted and natural uranium samples, and to less than 0.1% for the enriched uranium materials.

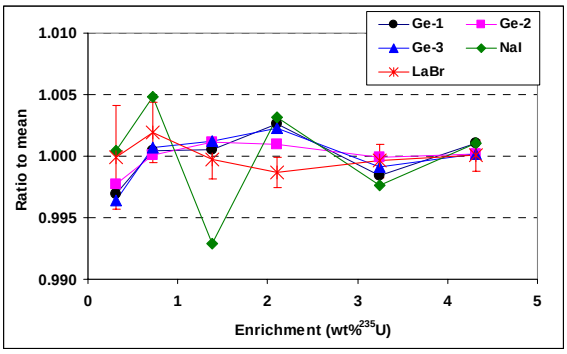


*Fig. 10. Normalized 186 keV count rate as a function of $^{235}U$ enrichment.*

The derived values for the corrected 186 keV count rates per mg $^{235}U$, relative to the mean value of the results for the 6 different enrichments, are plotted in Fig. 10. For the sake of clarity, error bars (1s) are only given for the data set obtained with the $LaBr_3$ detector. The same uncertainties also apply to the other data sets (in numbers: 0.32% for the depleted uranium sample, 0.18% for U-nat sample, and 0.13% for the enriched uranium samples). Numbers for the weighted mean values and the weighted standard deviations of the 6 measurement results per detector are given in Table 4.

From a visual inspection of Fig. 10, and from the derived values for the weighted standard deviations, we conclude: (i) the desired proportionality between 186 keV peak count rate and enrichment indeed

exists, and (ii) the performance of the $LaBr_3$ detector favourably compares with the performance of the HPGe detectors.

| Quantity | Ge-1 | Ge-2 | Ge-3 | NaI(Tl) | $LaBr_3$(Ce) |
|---|---|---|---|---|---|
| Self attenuation correction (% per change of 1 gU/l) | 0.063 | 0.068 | 0.063 | 0.055 | 0.038 |
| Weighted mean (186 keV counts/s per mg U-235) | 12.057 | 13.085 | 14.451 | 22.830 | 3.575 |
| Weighted standard deviation (%) | 0.124 | 0.0759 | 0.125 | 0.435 | 0.092 |

**Table 4.** Pertinent measurement data from linearity test.

Another aspect investigated separately concerns the question of measurement reproducibility for the counting configuration chosen for the $LaBr_3$ detector (Fig. 4). We had noted that the disposable polystyrene sample containers used in this measurement showed some variations in the bottom thickness of the order of up to 0.1 mm, which at the given sample–to-detector distance of 11 mm will notably affect the effective solid angle. For this reason only sample containers showing a uniform bottom thickness within ± 0.01 mm were selected for the enrichment measurements. The results of a reproducibility test, consisting of measurements on 7 different sample containers holding the same uranium solution, is shown in Fig. 11. Taking into account the average statistical counting error of 0.06 % per individual measurement, the results suggest a measurement reproducibility of about 0.12%.

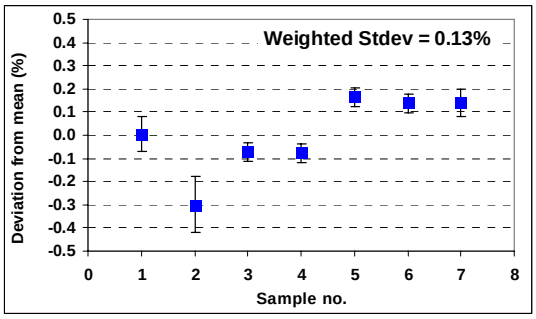


*Fig. 11. Results of reproducibility measurements on 7 samples of same enrichment.*

## 7. Outlook

We have demonstrated from our investigations that gamma spectrometry with a lanthanum bromide scintillation detector, if used in conjunction with the modified NaIGEM code for spectrum analysis, allows high-accuracy enrichment uranium measurements under the conditions of the COMPUCEA analyses. We are therefore planning to apply this type of detector in the 2nd generation of the COMPUCEA equipment, meeting then our initial design goal of having an instrumentation operating only with room-temperature detectors. Also with a $LaBr_3$ detector it appears realistic to keep the systematic measurement uncertainty for enrichment measurements at the level of 0.2% as set by the international target values for the present COMPUCEA measurements with a high-resolution HPGe detector [10].

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# Performance Evaluation of Genie 2000® Auto Gain Adjust

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***Abstract:***

*Auto Gain Adjust (AGA), a recently added feature in Canberra's Genie 2000® spectroscopy software, facilitates the consistent and automatic setting of a computer controlled amplifier gain to establish, maintain and monitor the accuracy of the existing energy calibration of a spectroscopy measurement system. The standard operation of AGA and the underlying methods employed are reviewed. Key inputs for the proper operation of AGA include a source certificate file, an automated analysis sequence, the preset live time for iterations, and the specification of the convergence criteria. The precision, consistency and limits of performance of AGA are evaluated. In particular, the AGA performance with HPGe and scintillation detectors using standard digital signal analyzers are assessed with recommendations for respective AGA parameter settings. Testing has been performed with a variety of check sources and with just background peaks. Measurement times range from <1 minute for laboratory counting systems using a standard check source to a few minutes when utilizing background energy lines. AGA provides consistent, unbiased precision with a final gain deviation from the initial value typically ranging from 1% to better than 0.05% depending on the energy resolution of the detector and the net peak area. Testing was conducted by manually simulating gain shifts of varying degrees in both directions prior to executing AGA. Automated quality assurance tracking is also supported. AGA is demonstrated to be a reliable new tool that avoids having to recalibrate while rigorously maintaining a desired system energy range.*

**Keywords:** automatic, gain, adjustment, spectroscopy, Genie

## 1. Introduction

Radiation spectroscopy systems require a way to maintain the accuracy of energy calibrations. In the Genie 2000 software v. 3.0 [1], the energy calibration process is simplified by the use of certificate files which are created with a graphical Certificate File Editor program. Certificate files contain information furnished with a calibration standard such as nuclide, emission rates, and energy lines. An initial calibration curve is obtained by matching this information to peaks found in the measured spectrum after the standard is counted. The energy calibration curve and the system gain $G_0$ at which the calibration was made are the reference parameters for the spectroscopic system. The accuracy of the calibration can be affected if a system gain shift occurs due, for example, to temperature fluctuations of the detector or the electronics when operating outdoors or in variable in-plant conditions, without concurrent digital stabilization.

The options we have in this case are to either recalibrate in energy every time a gain shift occurs or to adjust the electronics gain to the initial value $G_0$. Gain adjustment is preferred over energy recalibration because it resets the instrument to a standardized condition. Having a common energy calibration for a large number of spectra facilitates comparisons, consistency of analysis and following of trends. In this case, rather than having to recalibrate the whole system at the new gain value, it is useful to be able to execute a software-driven automatic gain adjustment (AGA) which would rematch the peaks in the spectrum to the existing energy calibration curve stored in the multi-channel analyzer input definition file for the detector.

The AGA procedure is used as a quality assurance (QA) tool between measurement sessions. Benefits of this method include automation and consistency in maintaining a desired energy range. Furthermore, the AGA

utility can be used periodically for checking, adjusting and tracking the gain during extended campaigns of system usage, when for example, the detector, the photomultiplier tube or the electronics present parameter drifts.

A gain evaluation portion of the AGA utility is used to evaluate the quality of peak energy calibration and as a tool for automated QA procedures.

## 2. AGA description and operation

The input parameters are specified and stored as part of the initial setting up of the utility. An A initially accurate energy calibration is a prerequisite for the AGA utility. The calibration curve is used to establish the correct location of the detected peaks. Before running the AGA algorithm, the optimum acquisition time for the given detector and source configuration must be determined, in order to obtain statistically significant peaks with relative errors less than 10%. This initial setup is required in order for the Analysis Sequence File (ASF) to correctly identify the reference energy lines above the peak significance threshold.

The AGA algorithm compares the reference energy lines specified in a certificate file with those in the measured spectrum and then it adjusts the gain of the amplifier in order to obtain peak matches within a specified tolerance. The AGA requires the following data inputs for use (Figure 1):

- Source certificate file defining the reference (true) energies intended to be measured during data acquisition.
- Peak analysis ASF that must include the Peak Locate – Unidentified 2$^{nd}$ Difference step and the Peak Area step.
- Live time preset for the duration of the acquisition iterations.
- Percent Gain Shift Tolerance (GST) to determine satisfactory convergence within a specified range around $G_0$.
- Iteration Limit to specify the maximum number of iterations.
- Optionally, a QA file with the data defined for export at the end of the AGA acquisition (such as peak centroid, peak energy resolution, etc.).

With this set of input parameters, the program acquires data over the first live time preset iteration and then performs the ASF in order to extract the found peak energies. The peak matching is made by comparing the measured spectrum with the true energies in the certificate file. For peak matching to be successful, at least half of the found peaks must be correlated to true energies, and vice versa. A new gain value is calculated so as to match the found peak energies with the existing energy calibration. The convergence criterion and the iteration limit are tested; if none of them are satisfied, the electronics gain is adjusted and new measurement iterations follow until the criteria are met.

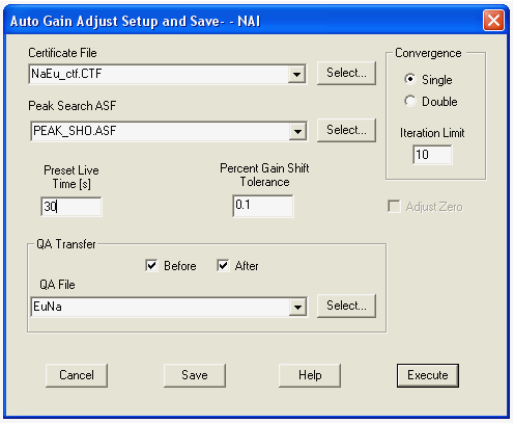


**Figure 1.** Auto Gain Adjust input screen example.

The system gain is a global parameter of the system, while the AGA algorithm runs iteratively with a number of energy lines. One criterion to establish the success of the gain adjustment is quantified by the gain deviation $\Delta$ from $G_0$, namely $\Delta = (G - G_0)/G_0 \times 100\%$. The algorithm runs successfully when $\Delta$ is within the limits set in the AGA input by the GST, such that $\Delta \leq GST$.

## 3. Testing the AGA performance

The precision, consistency and limits of performance of AGA have been evaluated with HPGe and NaI scintillation detectors using standard Canberra digital signal analyzers.

The precision in locating the peaks after a gain shift was quantified in units of Full Width Half-Maximum (FWHM) for each peak. The performance of the gain adjustment was quantified using the gain deviation $\Delta$.

Tests have been made with $^{22}$Na-$^{155}$Eu ISOCS source, $^{137}$Cs and natural background energy lines from $^{40}$K, $^{208}$Tl, $^{212}$Pb, $^{214}$Bi, and $^{228}$Ac. A certificate file has been created for each type of test source used for the AGA testing.

Before data acquisition, each detector has been calibrated in energy. In each case,

testing was conducted by manually simulating gain shifts of varying degrees from fractions of the initial system gain $G_0$ up to a factor of $5G_0$. Additional verification was made by running the AGA utility with no induced gain shift. The data presented in this work is extracted from the QA files and from the AGA run report.

### 3.1. Testing HPGe detectors with natural background energy lines

A set of 8 HPGe detectors with relative efficiencies close to 100% has been used to test the AGA with the natural background energy lines from $^{40}$K, $^{208}$Tl, $^{212}$Pb, $^{214}$Bi, and $^{228}$Ac. The data was individually acquired from each detector connected to a Canberra DSA 1000 multi-channel analyzer and a central computer.

The advantage of using the natural background lines is that is check source-free, but the disadvantage is that count rate can be rather low, on the order of 1-2 counts/s for the most intense line, in our case the 1460 keV line from $^{40}$K. Because the background was not known *a priori*, an in-house application named the Certificate Creator was developed in order to determine the energy lines that occur in the background spectra in more than 50% of the acquisitions. Twenty acquisitions for each detector were made with live time of 300 seconds. The data analysis was made in Genie with a nuclide library that includes the isotopes which are the most significant for the type of data being acquired. The reference (true) value of the found energy lines is then transferred to a new certificate file which can be used for AGA runs. Table 1 shows the most significant energy lines that occurred in more than 50% of the acquisitions, the net peak areas and their corresponding natural isotopes determined at the location of Canberra Industries in Meriden, CT. The six reliable lines have been subsequently used in the certificate file for AGA runs.

**Table 1:** Energy line occurrence frequency in the natural background for 300 s acquisition time.

| Energy line (keV) | % occurrence | Isotope |
|---|---|---|
| 238 | 75 | $^{212}$Pb |
| 352 | 81 | $^{214}$Pb |
| 609 | 91 | $^{214}$Bi |
| 910 | 69 | $^{228}$Ac |
| 1460 | 100 | $^{40}$K |
| 2614 | 68 | $^{208}$Tl |

AGA was run in parallel on the 8 detectors for a total of 200 times after manually changing the gain. The optimum acquisition time was of 300 s per iteration, with an average of 3 iterations required to adjust the gain. Successful gain adjustment was obtained for induced gain shifts ranging from 40% to 200% of the initial system gain, with GST = 0.1%. The precision in locating the peak centroid at the end of the AGA is quantified in Table 2 for the 352 keV, 1460 keV and 2614 keV energy lines, covering the typical energy range for the HPGe detectors. The % FWHM column shows the 1 standard deviation of peak centroid position in fractions of the measured FWHM.

**Table 2:** Peak matching results for the natural background energy lines for 300 s acquisition time

| True energy (keV) | Average measured energy (keV) | Net peak area (counts) | %FWHM |
|---|---|---|---|
| 351.93 | 352.05 | 67 ± 8 | 12.1 |
| 1460.83 | 1460.88 | 245 ±15 | 18.3 |
| 2614.53 | 2614.99 | 48 ± 6 | 14.6 |

Figure 2 shows the histogram of the peak centroid distribution for the 1460 keV line, with an energy bin of 0.25 keV. The data is summed for all 8 HPGe detectors. The standard deviation of the distribution is 0.36 keV.

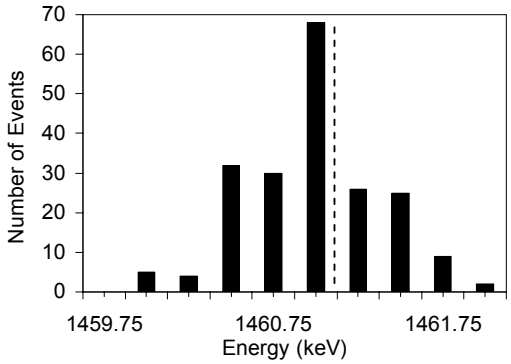


**Figure 2:** 1460 keV peak centroid distribution at the end of the AGA. The dotted line shows the position of the true reference energy (1460.8 keV).

For all the eight HPGe detectors used in this test, the AGA algorithm failed to adjust the amplifier gain when the shift was below 40% or above 200% from the initial gain value. These failures were caused by the inability to find at least half of the specified certificate lines. At extremely high gains, too many lines go high off scale. At extremely low gains, peaks become too bunched to discern.

### 3.2. Testing the HPGe detector with the $^{22}$Na-$^{155}$Eu ISOCS source

AGA tests have been run with a portable HPGe detector with 40% relative efficiency and

a 0.5 μCi $^{22}$Na-$^{155}$Eu ISOCS [2] source installed on the calibration jig at 10 cm from the detector. The standard ISOCS check source is convenient to use for the AGA utility for HPGe systems. The data was acquired with a Canberra InSpector 2000 DSA connected to a computer. The certificate for this measurement includes the 86.54 keV and 105.30 keV from $^{155}$Eu and 511.0 keV and 1274.53 keV from $^{22}$Na. The live time preset was 30 s per iteration, which resulted in net peak areas with relative errors less than 3.9%. The initial gain value at which the detector calibration has been made is $G_0$ = 23.35. Sixty-two AGA tests were run with an average number of 3 iterations per run in order to adjust the gain within 0.1% tolerance. A comparison of the true and measured energy lines is presented in Table 3. In this test, the %FWHM is at least a factor of two better than in the case of using the background lines, due to the better counting statistics resulted from the ISOCS check source.

**Table 3:** Peak matching results for the HPGe detector and ISOCS source.

| True energy (keV) | Average measured energy (keV) | Net peak area (counts) | % FWHM |
|---|---|---|---|
| 86.54 | 86.78 | 1572 ± 42 | 1.62 |
| 105.30 | 105.54 | 1190 ± 36 | 1.61 |
| 511.00 | 510.83 | 2360 ± 51 | 2.31 |
| 1274.53 | 1274.31 | 690 ± 27 | 7.61 |

Figure 3 shows the histogram of the peak centroid distribution for the 1274 keV line.

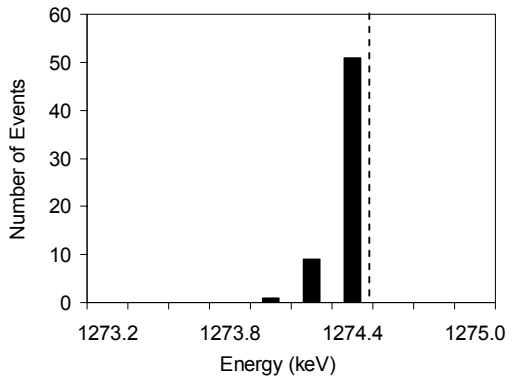


**Figure 3:** 1274 keV peak centroid distribution at the end of the AGA tests. The dotted line shows the position of the true reference energy (1274.5 keV).

The dynamic range of the gain shift was determined between $G_{min}$ = 2.25 and $G_{max}$ = 124 for the existing settings (from $0.1G_0$ to $5.3G_0$). An induced gain shift below $G_{min}$ and above $G_{max}$ value resulted in the failure of the AGA run, with the inability to find sufficient peaks.

The HPGe detectors tested with background lines and with the ISOCS source, resulted in a system gain restored with a deviation $\Delta < \pm 0.1\%$ from the initial value (Figure 4). Using the ISOCS source for HPGe detectors restores the system gain fast (under 2 minutes) and with $\Delta < 0.02\%$. By comparison, using the AGA with background lines result in a larger deviation than with a reliable ISOCS source. The use of background energy lines is justified when no check source is available and when the time to restore the system gain is not critical.

Both methods are entirely accurate enough for good quality analysis of the spectrum.

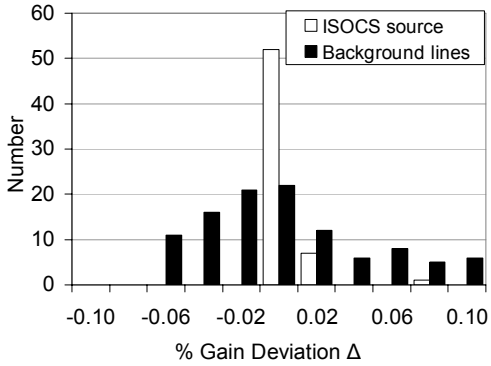


**Figure 4:** System gain deviation Δ from the initial value at the end of the AGA. Comparison for the HPGe detectors tested with the ISOCS source and with background energy lines.

### 3.3. Testing the NaI detector with $^{22}$Na-$^{155}$Eu ISOCS source and with $^{137}$Cs

In order to estimate the performance of the AGA feature with lower resolution detectors, tests have been run with a 2 inch NaI scintillation detector with a measured energy resolution of 6% at 662 keV. The data was acquired with a Canberra InSpector 2000 DSA connected to a computer. When using the 0.5 μCi $^{22}$Na-$^{155}$Eu ISOCS source, the optimum acquisition time was of 180 s per iteration. The gain shift tolerance (GST) was set at 1.0%.

A comparison of the true and measured energy lines is presented in Table 4 together with the peak statistics. The relative precision of the location of the peaks is better than for the HPGe detectors when expressed as %FWHM, due to the lower energy resolution of the NaI detector. As an example, the energy resolution at 1274 keV was 57 keV FWHM, whereas with a HPGe detector, the energy resolution was 1.97 keV FWHM.

**Table 4:** Peak matching results for the NaI detector and ISOCS source.

| True energy (keV) | Average measured energy (keV) | Net peak area (counts) | %FWHM |
|---|---|---|---|
| 86.54 | 86.45 | 12800 ± 131 | 1.82 |
| 105.30 | 105.95 | 8680 ± 112 | 1.78 |
| 511.00 | 510.90 | 15200 ± 131 | 1.85 |
| 1274.53 | 1274.23 | 938 ± 277 | 2.01 |

By comparison with the HPGe, the overall standard deviation of the NaI is a factor of 3 to 5 larger. The spread in the location of the 1274 keV peak is larger by a factor of 5 in NaI than in HPGe due to poorer energy resolution of the scintillation detector (Figure 5).

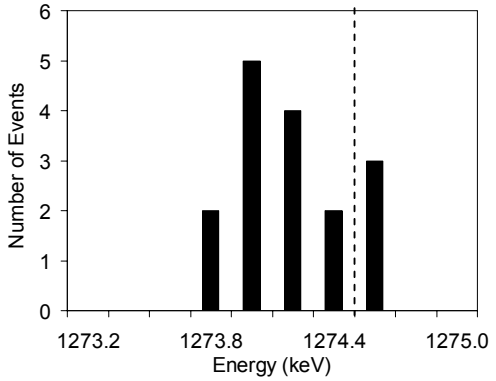


**Figure 5:** 1274 keV energy line hystogram with the NaI detector. The dotted line shows the position of the true reference energy (1274.5 keV).

With the initial calibration gain $G_0$ = 56.28, AGA was successful over a gain range of 7 to 95 gain units (12% to 168%). By comparison, when using the $^{137}$Cs source with a single reference line, the upper limit of the dynamic gain range was extended to 210 units (373%). Table 5 and Figure 6 present the statistics and the peak distribution frequency for the 662 keV line.

**Table 5:** 662 keV peak matching results for the NaI detector.

| True energy (keV) | Average measured energy (keV) | Net peak area (counts) | %FWHM |
|---|---|---|---|
| 662.00 | 661.97 | 13680 ± 116 | 1.36 |

When using the NaI detector, the gain was restored to within ±1% from the initial value (Figure 7). The larger tolerance in the gain value is consistent with the lower resolution of the NaI detector and the resulting imprecision in the peak location.

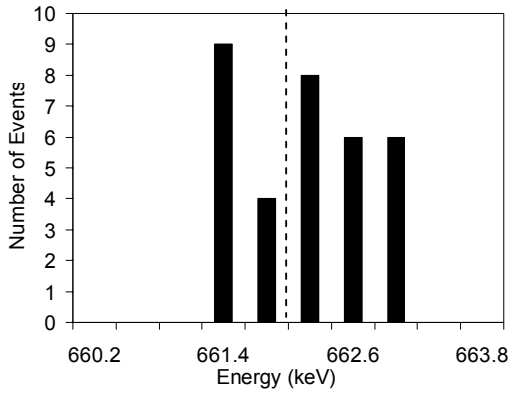


**Figure 6:** 662 keV energy line frequency distribution with NaI detector. The dotted line shows the position of the true reference energy (662.0 keV).

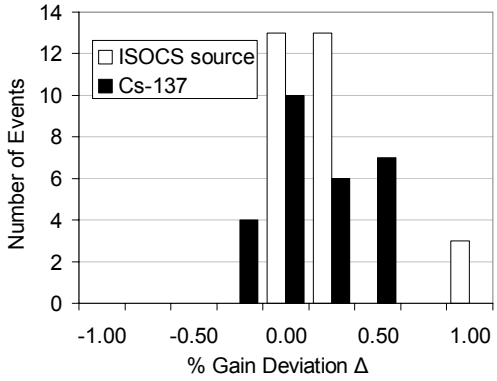


**Figure 7:** System gain deviation Δ from the initial value. Comparison for the NaI detector tested with the ISOCS source and with $^{137}$Cs.

## 5. Conclusions

The Auto Gain Adjust algorithm is a new QA tool implemented in Canberra's Genie 2000 ® version 3.0 radiation spectroscopy software. The tool was successfully run with HPGe and NaI detectors in order to evaluate operational limits for spectroscopy systems affected by variable operating conditions.

For each type of detector and check source configuration, the use of the AGA utility requires a one-time setup, with the live time optimized for the intensity of the source. After that, the AGA utility can be run on the spectroscopy system whenever it is necessary to adjust the system gain, without having to change the energy calibration.

The QA tracking embedded in the utility allows monitoring of peak parameters over time. The accuracy of gain restoration was better than ±0.1% for HPGe detectors and ±1% for the NaI detector. The choice of energy lines in the

reference certificate file is flexible, allowing for the use of background lines or check sources. Bidirectional gain shifts (to lower or higher values) have been consistently adjusted successfully between a factor of 0.1 and a factor of 5 from the initial gain value. The AGA utility can thus adjust for a wide range of gain shifts, including minor gain shifts around the initial gain value.

## 6. References

[1] Canberra Genie 2000 Operation Manual, http://www.canberra.com/products/834.asp
[2] Canberra ISOCS User Manual, http://www.canberra.com/products/838.asp

# Recent enhancements to the computer code MGA for the determination of the relative isotopic composition of plutonium from gamma-ray spectra

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***Abstract:***

*The Multi-Group Analysis code MGA is a highly regarded and widely used computer code for the analysis of high resolution gamma ray spectra in order to extract the relative isotopic composition of Pu for a diversity of items with minimal prior information. Accurate compositional analysis is required to quantitatively interpret correlated neutron and calorimetry assay measurements and also to establish nuclide specific activities for species which are difficult to quantify by direct peak area analysis. Over the past several years since the last major commercial release operational experience with the code has expanded and so too has the range of challenging measurement situations which the code is being asked to address. Detector technology has also advanced so that single planar detectors in the 100 $cm^3$ range have replaced small (<20 $cm^3$) and telescope combinations in a number of areas. This is far beyond what was envisioned by the original code designers.*

*In this paper we review the salient changes to the code that have taken place over the ver.9.6x sequence to the latest version which will be released as ver.10. A major objective has been to improve the robustness of the analysis for spectra with moderate to poor counting precision, for spectra of less than ideal resolution and for spectra with challenging combinations of radionuclides present.*

**Keywords:**MGA, relative isotopics code, gamma spectroscopy

## 1. Introduction:

The Multi-Group Analysis code MGA is a widely used gamma-ray analysis program for determining the isotopic abundances of plutonium and other actinides in a wide variety of samples. The MGA code was originally developed by Ray Gunnink and co-workers at the Lawrence Livermore National Laboratory [1], for use in the safeguards community using small volume planar Low Energy Germanium (LEGe) detectors. General overviews of the techniques used and results obtained from earlier versions of MGA may be found elsewhere [2-4]. The most recent version of MGA that is commercially available is the MGA ver.9.63H. This version of MGA dating back about 7 years was developed primarily for safeguards purposes and not intended, for example, to analyze spectra now being routinely obtained of waste assay measurements (i.e spectra with poor statistics, high attenuation, high $^{241}$Am, high $^{237}$Np, plutonium heat-sources, high fission products content etc.). A few years ago the code developers also received spectra of samples that were being analyzed incorrectly by MGA [5]. The new characteristic of these samples was their very high U/Pu abundance ratio. The drive to meet these challenges and a few suggestions received from various users for changes have resulted in substantial improvements. The new capabilities have been implemented in MGA ver.10. Most of the changes between earlier standalone versions of MGA and MGA ver.10 were described before at the recent international workshop on isotopic codes [4] , in this paper we want to discuss only the improvements since MGA ver.9.63H.

## 2. Code improvements made in MGA ver.10 over MGA ver.9.63H

Besides some changes to the user interface and the fact that the commercialised MGA ver.10 will support graphical outputs, e.g. it will be able to display peak fits and relative efficiency curve, a number of changes had been done to the computational part of the code. Some of the major code improvements over MGA ver.9.63H are described below. Numerous “trivial” housekeeping matters to do with switching compilers, for example, although important to the life cycle management of the code, are not of consequence to the user and will not be covered.

### 2.1 Analysis of Waste/Unusual spectra

Initially MGA was intended for use with spectra acquired using small volume planar Low Energy Germanium (LEGe) detectors. Due to their low sensitivity and poor efficiency at high energies LEGe detectors are more suitable for safeguards measurements and are often ineffective for general purpose radioactive waste characterization. Development of large volume Broad Energy Germanium (BEGe) detectors has given rise to increasing use of MGA in the area of nuclear waste assay [6]. BEGe detectors have a resolution at low energies equivalent to that of LEGe detectors, while at high energies the behavior is comparable to that of good quality coaxial detectors. The aspect ratio and large germanium volume make the BEGe a good choice for the efficient detection of special nuclear materials.

A series of test measurements performed at Canberra facilities and also results of routine in-field waste measurements showed that MGA ver.9.63H was, in many cases, not able to analyze certain waste spectra usually related to one of the following issues:

- hardware problems – e.g. pulse pileup, poor resolution, nonlinear amplifiers/ADC’s, see Fig.1. The detector resolution becomes one of the most critical parameters when analyzing a 100 keV region containing many overlaping peaks.

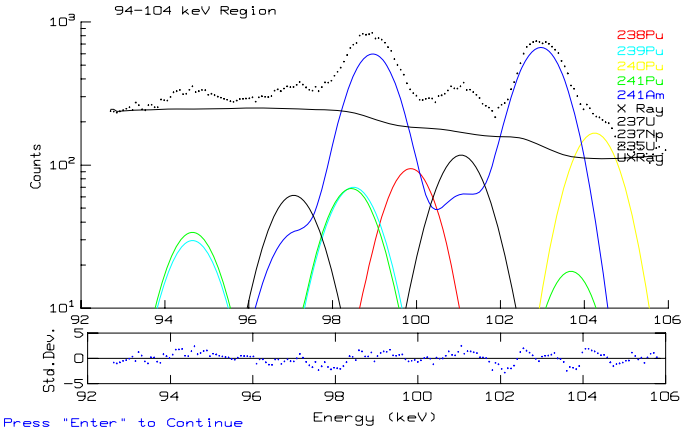


**Fig. 1** Example of the summed spectrum with poor resolution (FWHM 1326 eV at 122 keV)

- system configuration – poor setup and measurement technique
- source related problems – unusual isotopics, waste and scrap measurements, low count rate, high attenuation, high $^{241}$Am content, see Fig. 2 and Fig. 3. Spectra taken with high burn-up plutonium can often show a small 129 keV peak from $^{239}$Pu, and this may siginificantly affect the analysis results
- intrinsic limits – e.g. inability to quantify $^{242}$Pu (problem expecially with 2$^{nd}$ cycle plutonium where the necessary correlation functions are not established)

Failure of MGA to produce an output results in increased reliance on human expert analysis and default isotopics, which is costly, inconvenient and removes a source of confidence in the assay. To address this problem, we created an additional analysis option for “Waste/Unusual” spectra in MGA ver.10. When this analysis option is enabled the code has less strict internal requirements for the minimum number of counts in the spectrum analyzed and sets lower detection limits for $^{239}$Pu and $^{237}$U/$^{241}$Pu peaks. Although the resulting isotopic compositions and related information may be less accurate than is often considered in safeguard work, for waste it may be sufficient and additionally be of great help in the expert review process. During review any additional justification to adopt default data or confirm information can be valuable in difficult to measure cases. Instead of aborting, the code

now reports what it can, since this is far preferable to a failed report in such cases. By way of illusttation, even if $^{240}Pu_{eff}$ cannot be determined, a $^{241}Am/^{239}Pu$ estimate may still be useful in expert review by, for example, confirming the origin of the material.

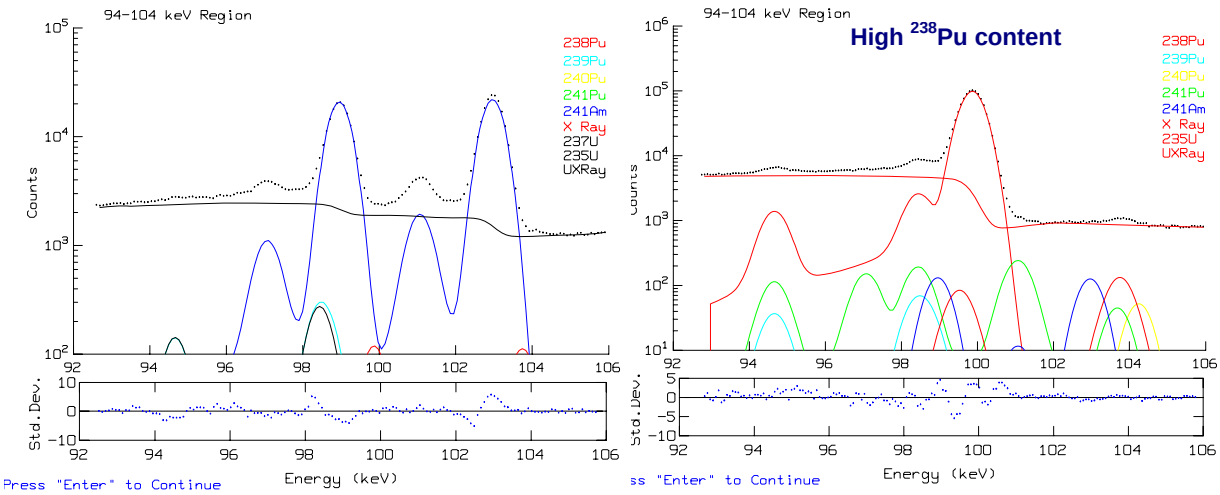


**Fig. 2** Examples of the spectra having unusual isotopic compositions

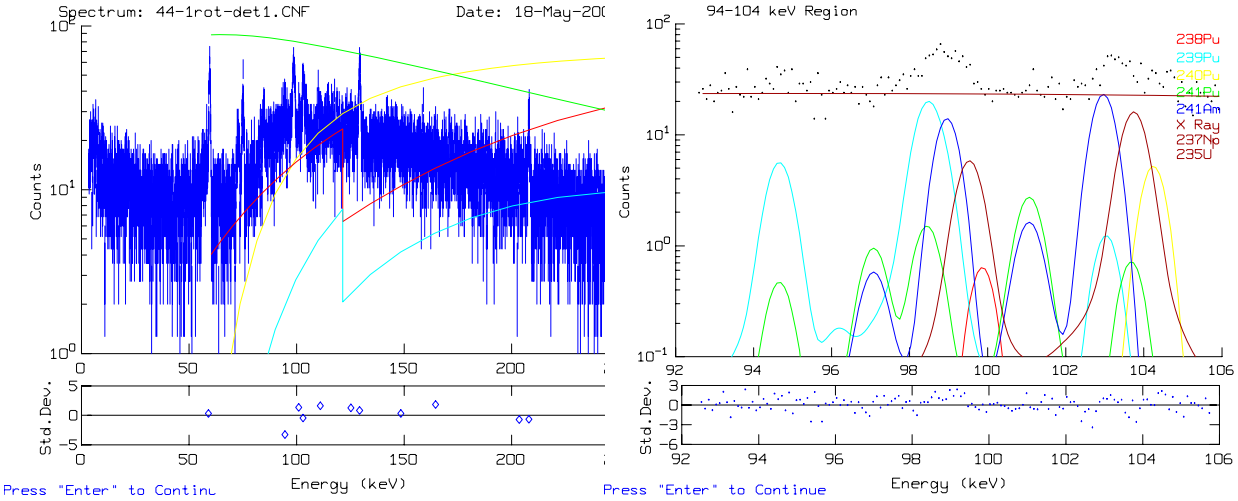


**Fig. 3** Examples of the spectra with low counting statistics

A series of measurements with reference plutonium samples have been carried out and then analyzed with both MGA versions [7] (sample isotopic range of $^{239}Pu$: 78 – 92%, $^{240}Pu$: 8-19%, Pu mass about 250 mg). Some of the experimental results for plutonium samples measured individually inside a 208 litre drum filled with polyethylene beads (density 0.6 g●cm$^{-3}$) are presented in Table 1. The different analyses are applied to the same spectra. MGA ver. 10 generally produces results that are close to the true value within the quoted uncertainties. It may be noted that for some spectra, especially for those with low counting statistics, MGA ver.9.63H results can significantly deviate from the reference data, when compared to MGA ver. 10.

Another set of data consists of the results representing in-field measurements [8]. In this case a number of real drums containing radioactive waste were assayed simultaneously with 4 BEGe detectors and the resulting summed spectra were then analyzed with MGA ver.9.63H. MGA ver.10 was only used when MGA ver.9.63H did not produce isotopics results or the results were not consistent with Acceptable Knowledge (AK). In many cases, when analyzing spectra having low counting statistics or high $^{241}Am$ interference, MGA ver.10 produced results that were close to the expected values (although the true values are of course not known). Such difficult to analyze spectra could not be properly treated by MGA ver.9.63H since the stringent internal tests were not met. Some of the experimental data are shown in Table 2. Where both versions produced results, they generally agreed well within reported precision, but fine adjustments made to the MGA ver.10 algorithms tend to produce values closer to expectations. This is to do with how peak overlap and interferences are treated.

| Sample ID | Measurement time, min | Meas./Decl. for $^{239}Pu$ | | Meas./Decl. for $^{240}Pu_{eff}$ | |
|---|---|---|---|---|---|
| | | MGA ver.10 | MGA ver.9.63H | MGA ver.10 | MGA ver.9.63H |
| CRM 136 | 30 | 0.99±0.02 | 0.94±0.02 | 1.07±0.14 | 1.38±0.16 |
| | 60 | 0.97±0.02 | 0.97±0.02 | 1.16±0.11 | 1.20±0.12 |
| | 90 | 0.98±0.01 | 0.98±0.01 | 1.10±0.09 | 1.12±0.09 |
| | 120 | 0.99±0.01 | 1.00±0.01 | 1.03±0.07 | 1.01±0.08 |
| CRM 137 | 30 | 0.99±0.03 | 1.17±0.03 | 1.02±0.11 | 0.36±0.12 |
| | 60 | 1.01±0.02 | 1.01±0.02 | 0.96±0.08 | 0.94±0.09 |
| | 90 | 1.03±0.02 | 1.01±0.02 | 0.89±0.06 | 0.94±0.07 |
| | 120 | 1.02±0.02 | 1.02±0.02 | 0.91±0.06 | 0.90±0.06 |
| CRM 138 | 30 | 1.00±0.01 | 1.00±0.02 | 1.00±0.15 | 0.98±0.23 |
| | 60 | 1.00±0.01 | 0.97±0.01 | 1.01±0.13 | 1.39±0.18 |
| | 90 | 1.00±0.01 | 1.00±0.01 | 1.03±0.09 | 1.06±0.13 |
| | 120 | 0.99±0.01 | 0.98±0.01 | 1.10±0.08 | 1.24±0.11 |

**Table 1** List of Measured-to-Declared ratios for $^{239}Pu$ and $^{240}Pu_{eff}$ for samples measured inside a 208 litre drum

| Count # | Meas./Expected. for $^{239}Pu$ | | Meas./Expected. for $^{240}Pu_{eff}$ | | Comments |
|---|---|---|---|---|---|
| | MGA ver.10 | MGA ver.9.63H | MGA ver.10 | MGA ver.9.63H | |
| 1 | 1.03 +/- 0.02 | no results | 0.51 +/- 0.25 | no results | Very low statistics |
| 2 | 1.01 +/- 0.01 | 1.01 +/- 0.01 | 0.93 +/- 0.09 | 0.83 +/- 0.09 | |
| 3 | 0.99 +/- 0.01 | 0.98 +/- 0.01 | 1.12 +/- 0.12 | 1.41 +/- 0.16 | |
| 4 | 1.00 +/- 0.01 | 0.97 +/- 0.01 | 1.04 +/- 0.10 | 1.48 +/- 0.17 | |
| 5 | 1.00 +/- 0.01 | 1.01 +/- 0.01 | 0.93 +/- 0.05 | 0.86 +/- 0.15 | |
| 6 | 1.00 +/- 0.01 | 1.01 +/- 0.01 | 1.01 +/- 0.05 | 0.78 +/- 0.09 | |
| 7 | 1.04 +/- 0.01 | no results | 0.41 +/- 0.15 | no results | Very low statistics |
| 8 | 1.00 +/- 0.01 | 0.94 +/- 0.02 | 0.99 +/- 0.01 | 2.02 +/- 0.38 | |
| 9 | 0.95 +/- 0.03 | 1.00 +/- 0.03 | 1.95 +/- 0.45 | 0.96 +/- 0.43 | Very high $^{241}Am$ |
| 10 | 0.97 +/- 0.01 | 0.99 +/- 0.03 | 1.55 +/- 0.19 | 1.10 +/- 0.52 | Very high $^{241}Am$ |
| 11 | 0.99 +/- 0.01 | 0.95 +/- 0.03 | 1.15 +/- 0.21 | 2.11 +/- 0.48 | Very high $^{241}Am$ |

**Table 2** List of Measured-to-AK ratios for $^{239}Pu$ and $^{240}Pu_{eff}$ for drums containing radioactive waste

## 2.2 U/Pu determination

There are now three methods in MGA for measuring the abundance of uranium in MOX samples [4]. Two methods were previously used in MGA ver.9.63H. In the first method, the $^{235}U$ content of a sample is measured by analyzing the 185 keV peak in the spectrum and the $^{238}U$ abundance is determined using the 1001 keV peak. However, there are some limiting conditions for this method. First, it requires a two-detector system or at least a detector system with an energy range extending to 1 MeV. Although single-detector systems having a range covering both the low- and high-energy regions can be used in principle, the user should remember that a large disparity exists (about a factor of 50) in branching intensities between the low- and high-energy gamma rays of uranium and plutonium. One advantage of the two-detector system is that Pb absorbers can be used to greatly attenuate the low-energy radiations, thereby allowing a closer source-to-detector configuration and corresponding improvement of counting efficiency at higher energies. Another problem is that the relative efficiency at 1001 keV must be extrapolated from regions that are 200-300 keV below it. It is felt that such a large extrapolation cannot be made to accuracies better than 3-5%. However, in its favor, this measurement approach can be used for a wide range of U/Pu ratios, as shown in Fig. 5 below. It is also amenable to waste, where broad energy range detectors are deployed and penetrating rays are needed to get results more representative for the volume.

The second method allows U/Pu ratio determination by measuring the self-induced fluorescence X-rays of U and Pu. Gamma rays above the K-shell binding energy of Pu will cause self-fluorescence in the sample, thereby generating K-shell X-rays in the spectrum. If the sample is MOX, additional U K-shell X-rays will also be produced. Whereas the only source of Pu X-rays is the self-fluorescence

process, U X-rays are also generated by internal conversion of some of the transitions associated with the decay of Pu isotopes. Both sets of X-rays experience considerable interference from surrounding X- and gamma radiations. Although the U/Pu ratio is obviously related to the measured intensities of the fluorescence X-rays, the relationship is not a linear one. First, there are small differences in the excitation coefficients of U and Pu which are, furthermore, also dependent on the isotopic characteristics of the sample. Second, radiations that have energies between the U and Pu K-shell binding energies (primarily due to Pu $K_\beta$ X-rays) can cause secondary fluorescence of the U in the sample but cannot fluoresce Pu.

The third method, implemented in MGA ver.10, provides correct analysis for samples with high uranium content and which have the 92-93 keV peaks of $^{238}U$ clearly observable in the spectra. Fig.4 gives an example for which these peaks are directly measurable. The code now automatically detects and measures the intensity of the 92-93 keV peaks and reports a U/Pu ratio based on this measurement. Our experience is limited to only one set of spectra. The results for this set are summarized in Table 3 along with results from the two other methods.

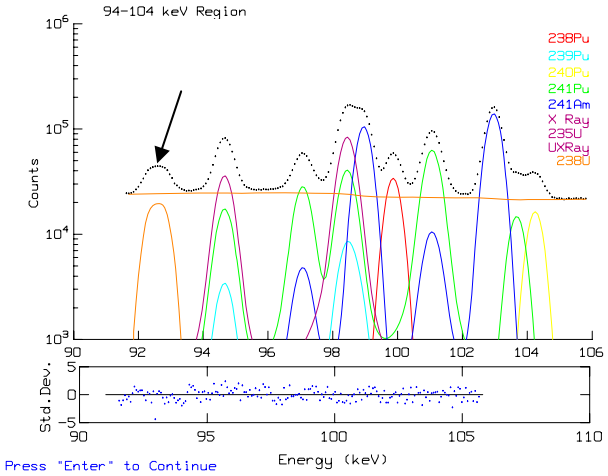


**Fig. 4** 92-93 keV peaks in high U/Pu spectra are clearly visible

| | No. of spectra | Range of U/Pu | Ave. Diff. (bias) | Std. Dev. |
|---|---|---|---|---|
| **186 – 1001 keV peaks** | 8 | 1.85 – 20.5 | -0.2 | 3.5% |
| **X ray fluorescence** | 17 | 1.85 – 20.5 | -0.3% | 2.3% |
| **92-93 keV peaks** | 14 | 15 – 50 | 0.7% | 2.0% |

**Table 3** U/Pu ratio results by method. Bias is against known composition

Each method has its range of greatest usefulness, as shown in Fig. 5.

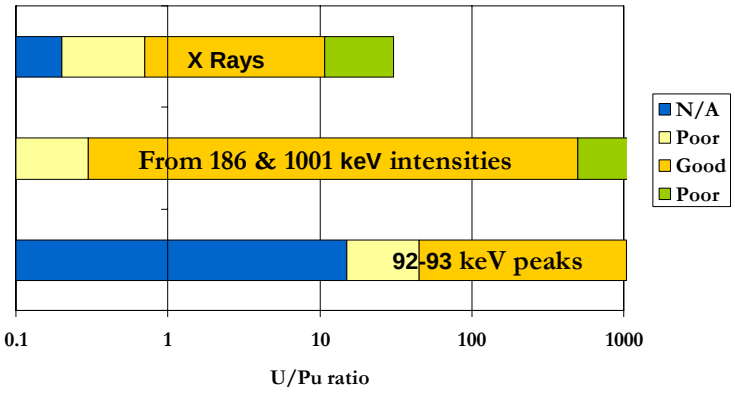


**Fig. 5** Ranges for good U/Pu measurements for each method

## 2.3 Flexible gain settings

For MGA ver.9.63H the energy calibration gain setting in the single low energy detector mode using 4k spectra must be approximately 0.075 keV/ch. In the two detector mode, the low energy spectrum gain setting must be approximately 0.075 keV/ch, and the high energy spectrum gain setting must be approximately 0.25 keV/ch. In the single high energy only mode using 4k spectra, the gain setting must also be about 0.25 keV/ch. For both spectra, slight deviations from the recommended gain

settings do not adversely effect the mathematical algorithms. However, large deviations from the nominal values may cause unpredictable results due to hard coded logic. For larger spectra (8k or 16k) MGA ver.9.63H allows somewhat greater flexibility in High Energy only mode when the Canberra Genie2k energy calibration is used.

However, a recent request was made to allow gains that would include the 300 and 400 keV regions in a 4k channel spectrum. This has led to the inclusion of flexible gain settings (user input or seeded from Genie2k). It should be noted that the 300 – 400 keV regions only become increasingly useful as the $^{239}$Pu content decreases, e.g. high-burnup samples. In these cases, the 129 keV peak becomes quite small and eventually may not be detectable. In such cases, the analyses can be greatly improved by including the 300-400 keV regions. Of course, a larger, yet very high-resolution, detector should be used so that a reasonable number of counts are obtained in these higher energy regions.

### 2.4 Analysis for highly nonlinear systems

It is still possible to encounter spectrum analyzers which exhibit extreme, non-acceptable, amounts of differential and/or integral nonlinearity. The gain (keV/ch) in the 100 keV energy region must be known precisely, e.g. an incorrect positioning of the 104 keV peak of $^{240}$Pu by a 0.1 channel can cause a considerable error in the result for this isotope. To counteract this potential problem, we have incorporated algorithms in an attempt to improve the analyses taken with highly nonlinear systems. Spectrum analysis problems of nonlinear systems are further exaggerated when the 59 keV peak is missing (due to highly absorbing materials) because the nonlinearity cannot then be detected or measured.

### 2.5 New detector efficiency algorithm

MGA is self calibrating for relative efficiency. An initial shape, based on a LEGe detector of user selected volume, is taken and the relative efficiency parameters are iterated to achieve a good fit to know peak ratios present for lines present in the spectrum. The original MGA algorithm for computing this estimated efficiency curve assumed that LEGe detector sizes would not exceed about 20 cm$^3$. To accommodate the much larger high-resolution detectors (BEGe) that are now available and routinely used, the detector efficiency algorithm has been expanded to estimate low-energy efficiency curves for detectors up to about 150 cm$^3$ in volume. This has proven not to be a critical addition because of the model used by MGA combined with iterative process. The difference in the relative efficiency shape depending on the detector size is illustrated on Fig. 6.

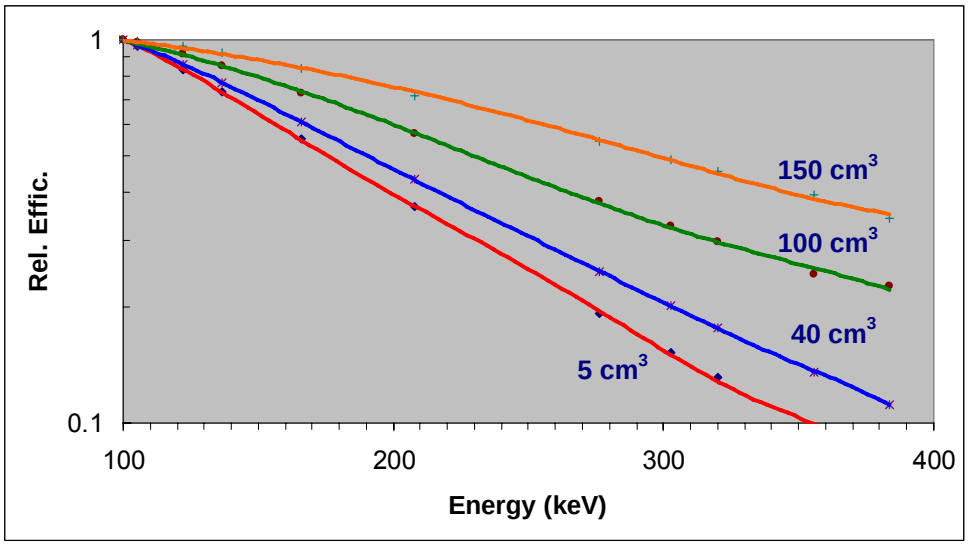


**Fig. 6** Relative efficiency for different detector sizes

## 3. Conclusions

Significant improvements to MGA have taken place in recent years and as expected the newest version of the code, MGA ver.10, was found to be more robust than previous versions. The previous version of this code, MGA ver.9.63H, developed several years ago and commercially available, was

primarily used in the safeguard community. Over the last years, especially after the new HPGe detectors combining a relatively large size of the crystal and a good resolution became available, the MGA code had to face new challenges. These included analysis of waste spectra, spectra with unusual and/or less than perfect properties such as having poor statistics, high attenuation, interference from fission products, etc… The drive to meet these challenges resulted in development of the improved version of MGA. MGA ver.10 still has all the capabilities available in the previous version of the code, but also contains a number of improvements:

- several operational modes: single low energy detector mode, single high energy detector mode, and two detector mode (where two detectors measure separate spectrums of the same sample)
- little or no user interaction
- automatically detects several sample characteristics requiring special treatment, e.g. MOX samples, fresh (non-equilibrium) samples, presence of $^{237}$Np and $^{239}$Np
- code has also been "hardened" to add a new capability of analyzing waste and unusual spectra
- less strict requirements to the counting system (gain flexibility, energy non-linearity)

MGA ver.10 has been extensively tested over past couple of years with various difficult spectra [7-8] as part of the evaluation process. These measurements showed a significant improvement in the code performance compared to the version V9.63H.

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# A Numerical Approach to Uranium Self-Absorption Correction using Lines from Multiple Isotopes

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***Abstract:***

*This paper extends a numerical method developed for Pu self-absorption correction (SAC) to non-destructive gamma spectrometry measurements of waste to apply to the case of uranium for special cases. If it is known that there are only a small number of possible uranium enrichment types that are present within a waste drum, and only one type is present in any given container, then it has been demonstrated that the apparent masses of different isotopes may be used to determine which enrichment is present. The SAC method used for Pu may then be applied using lines from different U isotopes.*

*A point kernel gamma-ray transport computer code has been used to model emission rates from U lumps with a range of masses from 0.001g to 350g and various shapes. A set of 3-Dimensional surfaces have been generated from the results for the true mass expressed in terms of the apparent masses obtained from the 186keV and 1001keV lines for different enrichment types (e.g. DU, natural and HEU). In the proposed U SAC method these surfaces are used to determine the true mass by a dedicated de-convolution technique. The surface to be used is selected by the determined enrichment.*

*The method has been tested against computer modelled data and against some measured data for HEU samples and the results are presented.*

**Keywords:** NDA, Waste, Self-Absorption, Uranium

## 1. Introduction

Although a significant amount of work has been performed on self-absorption of Pu gamma-rays in various materials, especially in relation to the differential attenuation of the viable lines, a standard correction technique for U SAC has not yet been developed. Recent work by Croft et al. [1] and Hansen [2] emphasise the interest in this problem and suggest that new and promising methods of U self-absorption correction (SAC) are currently being investigated e.g. by comparing the closely spaced X-ray lines from Th and U. If the sample may be identified as being between the extremes of 'dilute' or infinitely thick [3,4] then it may be possible to apply a pre-determined self-absorption factor (SAF) in order to obtain a compensated assay value.

Another possible method is to use the Infinite Energy Extrapolation (IEE) method [5], or a variation of it, whereby the apparent assay mass is plotted against the reciprocal of the energy, for example, and extrapolated to non absorbing conditions which is taken as 'infinite energy' on the basis that generally in the energy range of interest mass attenuation coefficients are falling rapidly with energy. This approach has been used successfully for Pu since $^{239}$Pu emits a number of lines of known abundance which are reasonably well spaced in energy. This may be adapted to use the 143keV, 163keV, 186keV and 205keV lines from $^{235}$U, however these lines are comparatively close together and do not offer as much differential absorption (sensitivity to lump size) as experienced by the 129keV and 414keV lines from $^{239}$Pu and also the dynamic range is limited by similar saturation points.

Furthermore the branching ratios are not as favourable. We use the IEE method as a comparison in section 5.

An alternative to estimating and applying a SAC from the measurement data of a particular assay could be to simply introduce an additional uncertainty component, a “lump” parameter for example. This component may be incorporated into the Total Measurement Uncertainty (TMU) associated with the measurement to account for the case where all the U in an item is concentrated into a single spherical (worst case) lump, depending on what may be known about the origin of the item. In so doing the assay value can be bounded in a way that may satisfy certain safety and transport criteria. But as a quantification approach it does not represent the best estimate value.

This paper extends a numerical method for Pu self-absorption [6] to apply to the case of U where the enrichment is known or at least falls into classes or bands. If only a limited number of possible enrichment types may be present within a waste stream and only one type is present in any given drum, then it has been demonstrated that the uncorrected characteristic lines may be used to determine which enrichment is present. This is possible if the source of the waste is known sufficiently well (e.g. from records or knowledge of the facility), however in principle the enrichment could be any value. Once the enrichment has been established the two-line SAC method developed for $^{239}$Pu may be applied. The two main lines used to measure uranium in waste drums are 186keV line from $^{235}$U and the 1001keV line from $^{238}$U (produced from the decay of $^{234}$Pa$_m$ as a second daughter of $^{238}$U if in equilibrium, which should always be the case for waste measurements). Although there are other $^{235}$U lines available (143.8keV, 163.3keV and 205.3kev), the 186keV line is the most prominent of the set, close enough to the U K-edge (115.6keV) to suffer a significant effect from photo-electric absorption for U and offers good differential absorption.

A program has been created to test a new U SAC method based upon the assumption that a waste stream contains only three possible enrichment categories. It has then been extended to perform with any known enrichment, which must be known either from details of the waste stream’s origin or using relative isotopic spectral analysis techniques such as employed in the MGAU or FRAM codes.

## 2. U Modelling

A point-source point-detector model has been used to model the behaviour of single unencapsulated U lumps. The model divides the lump up into a specified number of voxels (volume elements), defined by radial, polar and height integration intervals for cylindrical lumps, and lateral, depth, height and rotation integration-intervals for cuboidal lumps.

The model sets the gamma-ray production in each voxel as the volume of the voxel and computes the probability of the gamma-rays reaching the detector without scattering or absorption – both processes remove the line from the detected spectral peak. The voxel contributions are integrated over the whole lump and the calculation repeated with the material density in the attenuation function set to zero, the ratio of these results giving the SAF (self-attenuation factor) for that gamma-ray line. The scattered component of the gamma-ray flux reaching the detector can safely be ignored when high resolution Ge detectors are used in the measurement of the 186keV and 1001keV lines since the scattered photons are almost invariably energy degraded to the continuum below the photo-peak.

It shall be assumed here for illustrative purposes that there are only three possible enrichment categories within waste drums. Although real waste may have different enrichments, modelling and correction may be performed for any U enrichment value, provided the possible enrichments can be grouped into discrete values or bands from prior knowledge of the waste types present. Three commonly used enrichments have been chosen to demonstrate the way enrichments can be estimated from apparent enrichments and consequently fed into a U SAC. The three chosen enrichment types for the purpose of the investigation are: Depleted uranium (DU) 0.5% $^{235}$U, Intermediate enrichment uranium (IEU) 40% $^{235}$U and High enrichment uranium (HEU) 93% $^{235}$U.

The self-absorption curves for U (of density 18.7g.cm$^{-3}$) have been modelled to determine the SAF and hence apparent masses at the 186keV and 1001keV lines for the three possible enrichment types: DU, IEU and HEU. The uncorrected self-absorption behaviour of bare U lumps of various

dimensions has been calculated using the point kernel model, over the mass range of 0.1mg to 350g for the following geometries:

- Cylinders with diameter equal to height, with dimensions ranging from $2.0 \times 10^{-4}$m to $2.9 \times 10^{-2}$m
- Square faced plates with a fixed depth of 1mm, with heights ranging from $7.0 \times 10^{-5}$m to $1.6 \times 10^{-3}$m
- Rods with a fixed length of 0.1m, with diameters ranging from $8.3 \times 10^{-6}$m to $1.5 \times 10^{-2}$m
- Cubes of side length ranging from $2.0 \times 10^{-4}$m to $2.7 \times 10^{-2}$m
- Plates of fixed square face length $3.0 \times 10^{-3}$m, with heights ranging from $6.0 \times 10^{-7}$m to $5.9 \times 10^{-3}$m
- Plates of fixed square face length $3 \times 10^{-2}$m, with heights ranging from $6.0 \times 10^{-7}$m to $2.1 \times 10^{-2}$m
- Plates of fixed square face length 0.1m, with heights ranging from $5.0 \times 10^{-7}$m to $1.9 \times 10^{-3}$m
- Plates of fixed square face length $7 \times 10^{-2}$m, with heights ranging from $5.0 \times 10^{-7}$m to $3.8 \times 10^{-3}$m
- 6cm high cylinders, with diameters ranging from $1.1 \times 10^{-5}$m to $2.0 \times 10^{-2}$m
- 1cm high cylinders, with diameters ranging from $2.6 \times 10^{-5}$m to $4.9 \times 10^{-2}$m
- Square faced plates with a fixed depth of 1mm, with heights ranging from $4.0 \times 10^{-5}$m to $7.9 \times 10^{-2}$m
- Zero attenuation geometry (186keV mass = 1001keV mass = true mass)

## 2.1. Results

Figure 1 shows the relationship of apparent masses at 186keV ($M_1$) and 1001keV ($M_2$) and the true U mass ($M_T$) for the selection of shapes for three enrichment types. The DU, IEU and HEU sets are labelled and although the 3-D surfaces for the latter pair are very close together they do not overlap. This may not be the case for other sets of enrichment types however.

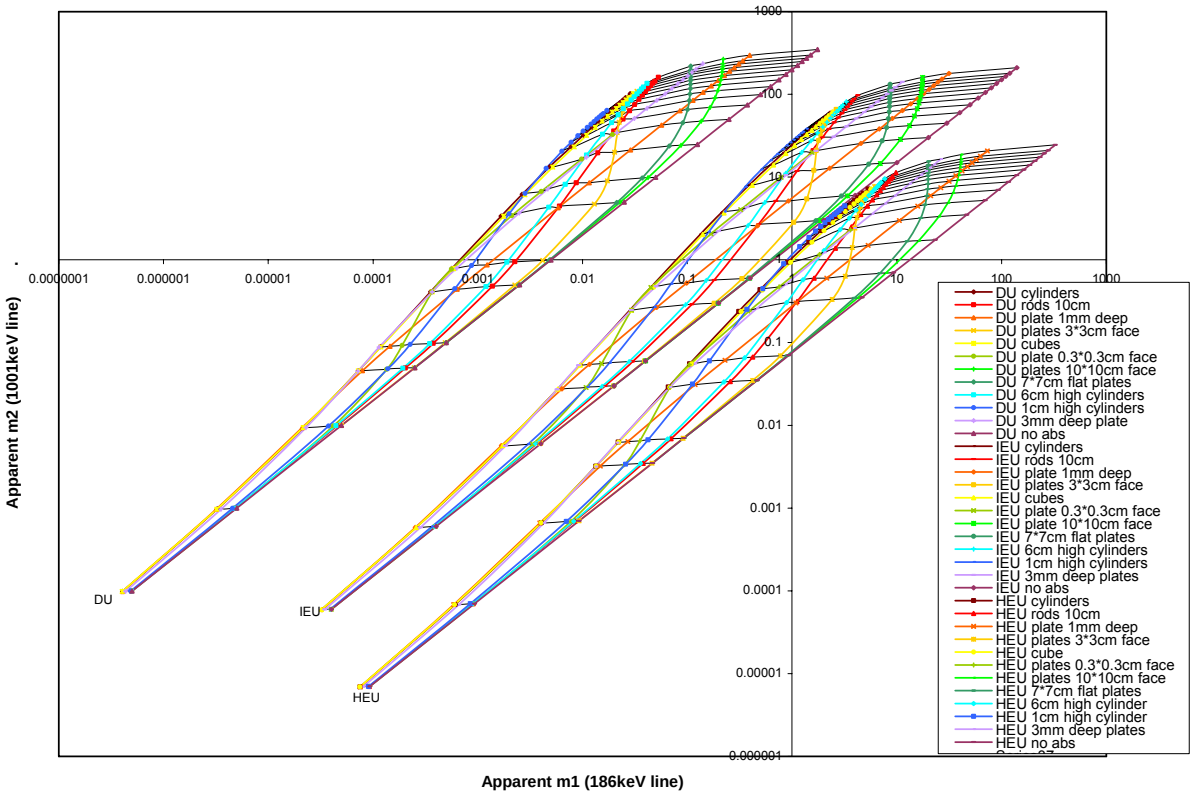


**Figure 1:** Uranium self-absorption for three specified enrichment categories.

Each set of curves follows a similar pattern to those seen for Pu as the U becomes more self-absorbing. The near-horizontal black curves are a selection of constant true mass lines. The diagonal lines (for which $M_2 = kM_1$) are the cases of no attenuation where $M_1$ divided by the $^{235}U$ enrichment is equal to $M_2$ divided by the $^{238}U$ enrichment. Very small mass lumps will lie on or close to the no

absorption lines, and as the lump size increases and the shapes become more attenuated (for the 186keV line), the curves migrate towards the opposite upper edge of the surfaces. As the mass and lump size increase further they saturate and the points on the curve bunch closer together. No points should lie outside the respective surfaces for the relevant enrichment type.

Based on this database of calculated results a U SAC method (a software engine) has been developed to calculate the mass of U from the two apparent masses $M_1$ and $M_2$, and has been used to demonstrate the effectiveness of the new self-absorption technique. The new method is described briefly below.

## 3. The USAC Algorithm

A Borland™ Delphi program has been created to calculate the enrichment based upon the inputted apparent masses (these are the assay results fully corrected for matrix attenuation in the drum and other factors as best as we are able). The apparent enrichment $E_{app}$ may be calculated as:

$$E_{app}(\%) = \frac{m_{U235}(g)}{m_{U235} + m_{U238}(g)} \times 100$$

where $m_{U235}$ is the apparent mass of $^{235}U$ from the 186keV line and $m_{U238}$ is the apparent mass of $^{238}U$ from the 1001keV line. Due to self-absorption the apparent masses will always be less than the true mass and the less energetic line will always be affected more. Consequently the apparent enrichment should be less than the true enrichment (although this may not strictly be the case due to statistical uncertainties in the measurement) where self absorption is significant, and the apparent enrichment may be used to select an enrichment category, provided the regions on the apparent 186keV mass versus apparent 1001keV mass surface do not overlap for the different enrichments.

Using this enrichment category the program will scale the $M_1$ (apparent mass from the 186keV line), $M_2$ (apparent mass from the 1001keV line) and $M_T$ (the true mass) values accordingly and determine cubic spline fits for each enrichment and shape combination, plotting curves for $M_T$ versus $M_2$, and $M_1$ versus $M_2$ to obtain the self-absorption corrected mass and $M_1$ as functions of $M_2$. Using the calculated cubic spline fits and the measured 1001keV line apparent mass ($M_2$) we may take a slice of the 3-D surface with varying true mass and $M_1$ and plot the ($M_1$, $M_T$) pairs calculated. If the locus between successive points may be approximated as a straight line the SAC U mass may be determined from the measured 186keV line apparent mass ($M_1$).

## 4. Modelled Results

The USAC algorithm has been tested with modelled data for three new lump types with a mass range of 0.1mg to 350g. These lumps are 15mm high cylinders, 2mm fixed depth flat plates and 40mm high cylinders. It is necessary to ‘test’ the method against modelled results for shapes etc that are **not** included in the base data set or ‘incestuously’ over-optimistic results may be obtained.

The modelled data for each set have been converted for DU, IEU and HEU and run through the U SAC engine. In each case the engine chose the enrichment type correctly. The results are shown in Figure 2.

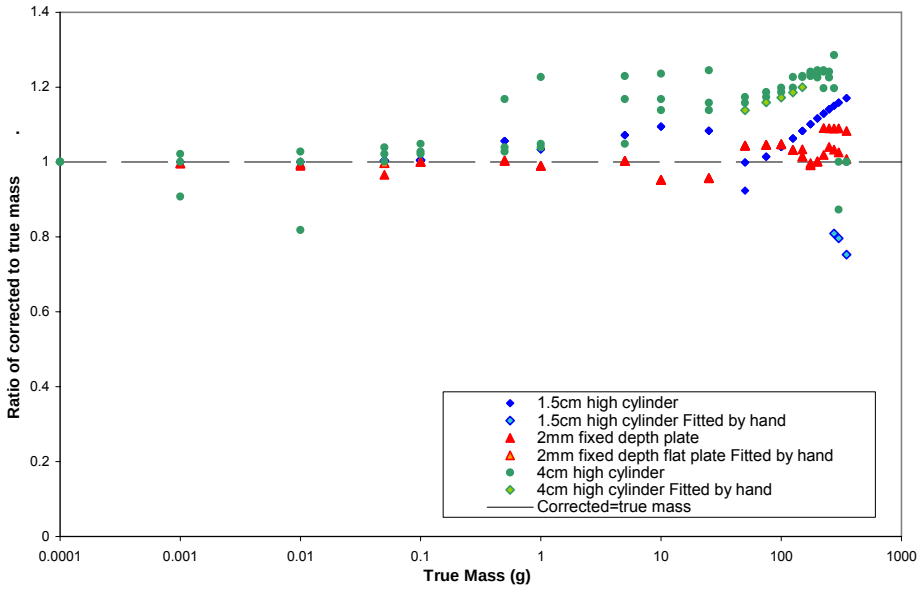


**Figure 2:** Graph showing the calculated to true mass ratio using the developed U SAC method.

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For each new lump type modelled, there are cases where the engine begins to underestimate the true mass result since the data points fall just outside the 3-D surface and the linear least squares extrapolation method is used to determine the true mass. Since the data points in the high attenuation region are clustered the least squares extrapolation does not offer a good fit to the whole curve.

The results for the 2mm deep plates all fall within 10% of the true mass across the range of masses. The results for the 40mm high cylinders have been found to overcorrect at higher masses, and the results for the 15mm high cylinders begin to overestimate with increasing mass above 100g. Overall the USAC engine appears to work well and the results are within approximately 25% of the true mass which is impressive in the context of the bias that would result without compensation.

## 5. Experimental Results

The program outlined in the previous section has been extended to be applicable for any possible enrichment. The enrichment must be stated by the user prior to its execution and so must be determined externally. Where the enrichment as been determined from the spectral data and cannot be categorised into a discrete enrichment type, there will be an additional uncertainty associated with the $^{235}$U and $^{238}$U isotopic estimations. For these cases the uncertainty in the total U SAC mass has been determined from the standard method for combining uncertainties from multiple independent variables by:

$$\sigma_{M_T}{}^2 = \left(\frac{\partial M_T}{\partial M_1}.\sigma M_1\right)^2 + \left(\frac{\partial M_T}{\partial M_2}.\sigma M_2\right)^2 + \left(\frac{\partial M_T}{\partial En_{235}}.\sigma En_{235}\right)^2 + \left(\frac{\partial M_T}{\partial En_{238}}.\sigma En_{238}\right)^2 \text{ (g)}$$

where $En_i$ is the estimated enrichment of the sample for isotope *i*, $\partial M_1 = 0.001 \times M_1$, $\partial M_2 = 0.001 \times M_2$, $\partial En_i = 0.001 \times En_i$, $\partial M_T$ in each case are the changes in $M_T$ when $\sigma M_T$ is mapped through the USAC process, $\sigma En_i$ is the uncertainty in the enrichment for isotope *i*, and $\sigma M_n$ are the inputted uncertainties in the 186keV or 1001keV masses at the one standard deviation level providing the total uncertainty in $M_T$ is at one standard deviation.

The USAC code has been tested using U 'Zebra plates' (sections of fuel fabricated for an experimental reactor) with an enrichment of approximately 93%. The source thickness has been

estimated based upon the known mass and areas of the plates and an assumed density of 18.7g.cm$^{-3}$ for U metal.

**Table 1:** Uranium Zebra plate data including total U masses and plate dimensions.

| ID | U mass (g) | U-235 mass (g) | Dimensions (mm) |
|---|---|---|---|
| 4005 | 17.91 | 16.66 | 25 x 25 x 1.5 |
| 4006 | 17.94 | 16.68 | 25 x 25 x 1.5 |
| 4007 | 17.99 | 16.73 | 25 x 25 x 1.5 |
| 4008 | 17.95 | 16.69 | 25 x 25 x 1.5 |
| 4009 | 9.56 | 8.89 | 25 x 13.5 x 1.5 |
| 4010 | 4.21 | 3.91 | 11 x 14.0 x 1.5 |

The sources have been measured using a 20mm thick, 70mm diameter BeGe detector, with a polyethylene end cap, at a distance of 100mm from the end cap, in the following combinations:

- 4010 mass = 4.21g
- 4009 mass = 9.56g
- 4008 mass = 17.95g
- 4008-9 mass = 27.51g
- 4007-8 mass = 35.94g
- 4007-9 mass = 45.5g
- 4006-8 mass = 53.88g
- 4006-9 mass = 63.44g
- 4005-8 mass = 71.79g
- 4005-9 mass = 81.35g

Multiple sources have been stacked behind one another, positioned vertically with the largest face of each towards the detector and with the smaller sources in the group closest to the detector. The apparent masses for the 143.8keV, 163.3keV, 185.7keV, 205.3keV and 1001keV lines have been determined and used with the USAC program for non-discrete enrichments. The data have also been analysed according to the Infinite Energy Extrapolation method with both linear and polynomial fits being applied to carry out the extrapolation to 1/E=0. The results are shown in Figure 3.

The sources have been measured statically (non-rotating) with the face of the source approximately parallel to the face of the detector, whereas the developed U SAC program has been designed to work with lumps which present a changing orientation to the detector (as they are moved along with the contents of the drum as it is rotated in front of the detector and also as the drum is viewed from several vertical positions) and so the apparent masses at all energies are larger than expected for a rotated geometry, with a stronger effect experienced by the lower energy lines. However, the corrected result using the USAC method is more dependent on the change in 1001keV apparent mass and so the overall effect is the method still performs reasonably well. In a sense the 186keV line is indicative of the projected surface area for reasonably large lumps and the 1001keV is more of a bulk indicator in this regime.

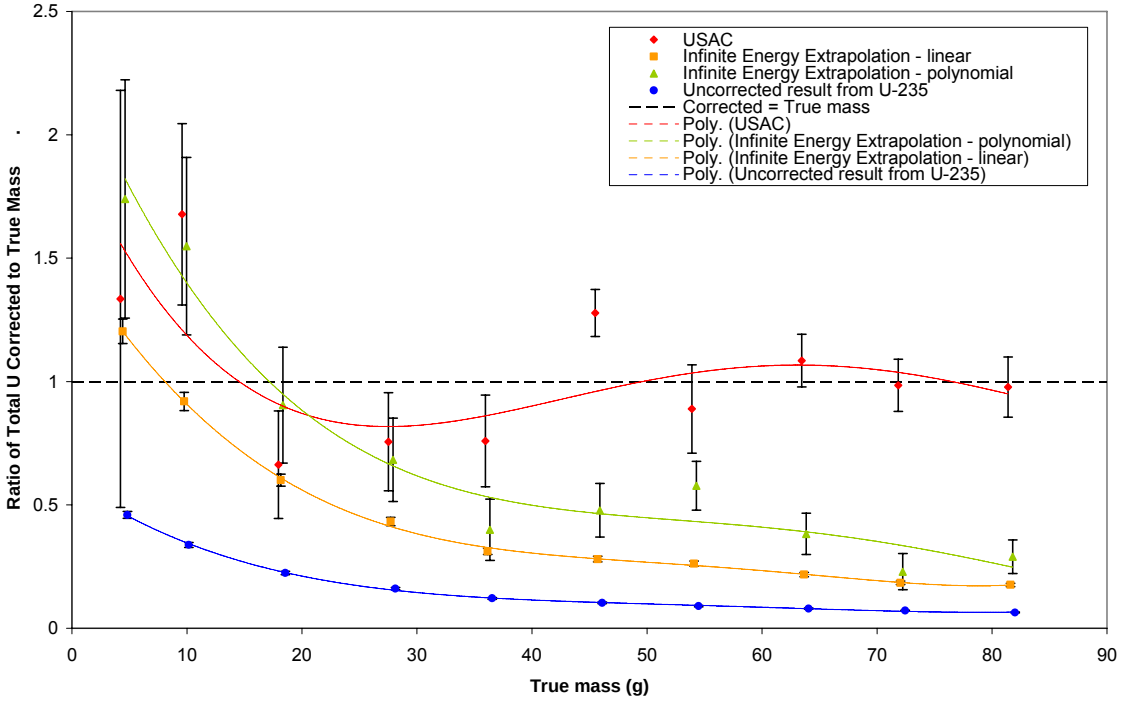


**Figure 3:** The ratio of corrected to true total U mass as a function of true mass for U Zebra plate measurements showing USAC and Infinite Energy Extrapolation correction methods

Figure 3 shows the total U results calculated with the USAC method described. The uncorrected results begin at half of the true mass and reduce continuously as the mass increases. Both IEE fitting methods produce results which reduce to below 0.3 of the true mass for higher massed samples. This may be because the line energies used for the fitting methods are small and so there is not a high energy (less attenuated) line to improve the extrapolation to 1/E = 0, resulting in a lower U mass from these more attenuated lines.

The USAC algorithm has been shown to perform very well, with the majority of results within 35% of the true mass, offering a significant improvement compared to the other methods. The largest deviations from the true mass have most likely occurred mainly due to statistics since HEU produces very few 1001keV counts, as indicated by the large error bars for these measurements. The sources were measured for live times of between 600s and 1200s depending on the mass of the sample, whereas in real assay systems, especially with heavier matrices, the practical counting time may be too short to obtain enough 1001keV counts for HEU unless the U mass is very large. However, for lower enrichment U samples (with a higher $^{238}U$ content) would provide improved statistical accuracy.

One advantage to the USAC method is that it may be used for mixed U and Pu drums without difficulty since it uses only the 186keV and 1001keV lines, whereas the IEE methods and the X-ray fluorescence methods can both suffer from interference from Pu lines.

## 6. Extension to Multiple Lumps

The extension to waste packages that contain U in the form of multiple lumps (which may be of different sizes, shapes and materials) has been investigated. In practice it is rare indeed for a waste item to contain only a single piece of fissile material. Due to the high degree of non-linearity observed in the model, the combination of a series of smaller lumps was expected to produce a smaller self attenuation factor than that expected for a single large lump of equal mass and therefore the model was expected to underestimate the total U in a sample. Thus far we have considered the impact on DU result only but due to the nature of the algorithm the other enrichment results ought to behave similarly.

The apparent 186keV and 1001keV masses for a number of material/shape combinations were summed and used as input for the algorithm to see how it would behave for multiple lumps. These combinations include several lumps of the same material and shape, plus random mixed numbers and types of lumps over the range of masses and shapes studied in the USAC algorithm development, plus the series of newly modelled data sets used for the modelled results. The results are presented in Figure 4.

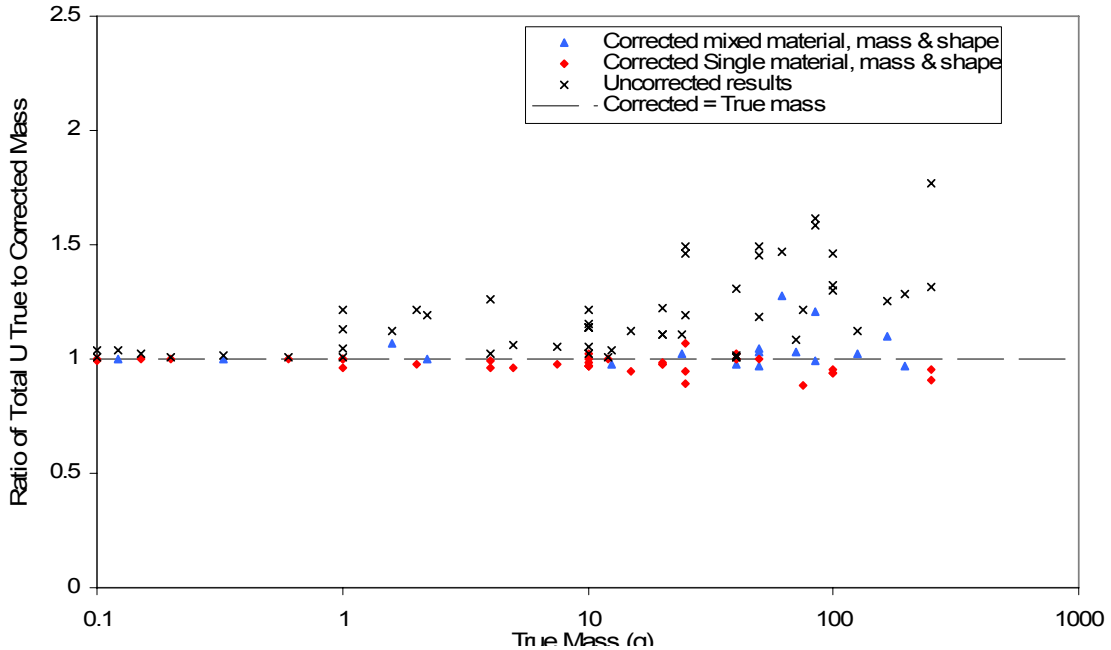


**Figure 4:** Plot of the ratio of true to corrected total U mass versus the total true mass for random combinations of lumps collated from independently-treated single lump modelled data for DU.

The uncorrected total U mass (derived form the186keV apparent masses and isotopics) are also plotted as a comparison. The results do not produce perfect ratios of one, as expected, because as the individual lumps are combined their total 186keV apparent mass will be higher than that of a single lump of equivalent mass, whereas the 1001keV apparent mass will remain relatively similar. Consequently the corrected mass of the combination of lumps is underestimated, demonstrated by approximately 70% of the USAC corrected data points. The remaining corrected masses lie above the ratio equals one. This is line due to fluctuations in the data points in the high attenuation region of the $M_1$:$M_2$:$M_T$ plot.

It should be noted that these lump combinations have been performed by summing the apparent masses of individual lumps, assuming that all lumps are independent and the gamma rays from each lump are not attenuated by any other lump on their path to the detector. However this may not be the case in real waste drums and the resulting apparent mass may, conceivably at least, be attenuated more than for the examples shown above.

The method appears to work well for multiple lumps of various compositions. All of the modelled results are within 28% of the corrected equal to true mass line, with 79% of the modelled results achieving values within 5% of the true mass. These observations could be developed with a more detailed study to implement a contribution to the TMU of the sample from the U self-absorption in the case of multiple lumps. For example, if a 1-σ uncertainty is required by the user (68% of the results should lie within the area if a Gaussian distribution applies, for example), this equates to a 3.5% uncertainty if the uncertainty is assumed to be constant across the mass range.

It may also be observed that the variation from the corrected equal to true mass line appears to increase with increasing true mass and so the TMU contribution from multiple lumps could be refined to be a function of the apparent mass with a more in-depth study.

## 7. Conclusions

In this work we have re-examined how the self absorption correction may be derived and have extended a Pu self-absorption correction method to the case of U using the 186keV and 1001keV lines for those cases where the $^{235}U$:$^{238}U$ ratio is known or can be determined. An empirical approach was adopted where the nature of the empirical study involved calculating the behaviour of a large range of lump types (shapes, compounds and densities) and letting nature reveal the correlations with promise. The method has been tested using both modelled and measured data.

The correction method has been applied to lumps that are uniform and is based on the use of a single pair of lines. In practice the majority of drums will contain multiple lumps with a distribution of lump sizes. A simple study of multiple lumps has been presented and has been found to perform satisfactorily.

The method may be applied to real waste streams to provide a very significant improvement in U mass determination compared to gamma-ray assay in which no self-absorption method is being applied. The method does require either that the enrichment of the sample is known from details of the waste stream or via spectrum analysis programs such as FRAM or MGAU, or if this is not possible, the enrichment may be determined from the apparent mass data provided the enrichment may be selected from discrete enrichment categories – iteratively if need be.

The USAC engine has been incorporated into an assay system and we look forward to reporting future results of plant performance. In addition the TMU estimation and performance estimates for lump distributions are being refined.

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# Implementation of an experimental design to evaluate the codes used to determine the enrichment of uranium samples

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## Abstract

Gamma spectroscopy is commonly used in nuclear safeguards to measure the enrichment of uranium applying either the enrichment meter method, or analysis codes developed to analyse several $\gamma$-rays or X- and $\gamma$-rays in defined regions of a uranium spectrum. An experimental design has been carried out for the measurement of uranium enrichment using this technique with different software used in safeguards applications: MGA, MGA++, PCFRAM and IGA. The main goal is to improve the estimates upon the uncertainties of the measurement taking into account realistic in situ acquisition condition.

To set up the experiment, uranium sources have been achieved. It is $U_3 0_8$ reference material in powder form, with a wide enrichment range in sealed container. Regarding the acquisition systems, several gamma spectroscopic germanium detector (planar, coaxial) and multichannel analysers have been used. Five entry parameters have been considered for the experimental design: $^{235}U$ enrichment, adjustment of the acquisition gain, source-detector distance, shielding, and matrix effect. With all these spectra, there is a direct comparison between the uncertainty announced by the different software and the difference between the computed enrichment value and the value of the reference material. The paper describes the methodology implemented to evaluate the results of such an experimental design, and exhibits the results obtained when applying the code MGAU, with the objective to confirm or to improve our current knowledge of uncertainties for these uranium enrichment measurements. The evaluation of the 3 other codes, which is currently in progress, will be subsequently published.

## 1. Introduction

Gamma spectroscopy is commonly used in nuclear safeguards to measure the enrichment of uranium samples applying either (1) the traditional enrichment meter method based on the counting of the 185.7 keV $\gamma$-ray of $^{235}U$, or (2) analysis codes developed to analyse several $\gamma$-rays, or X- and $\gamma$-rays present in defined regions of the acquired uranium spectrum, without the need for any calibration. The isotopic composition analysis codes MGA [1], MGA++ [2], PCFRAM [3] or IGA [4], which analyze $\gamma$ emission spectra of uranium or plutonium, produce results accompanied by an uncertainty whose dominating component has a statistical origin. The experience feedback at IRSN as well as the conclusions reported in [5] and [6] had emphasized situations of measurement where calculated uncertainties are not representative of the real bias made by the code. To address this concern, IRSN has carried out an experimental design for the measurement of uranium enrichment with the previous analysis codes in order to better estimate the measurement uncertainties taking into account realistic in situ acquisition conditions.

The paper first reports on the experimental setup, describing the characteristics of uranium reference samples especially achieved, the implemented acquisition systems, the parameters of the experimental design ($^{235}U$ enrichment, adjustment of the acquisition gain, source-detector distance, shielding, and matrix effect), and the measurement procedure. With all these spectra, there is a direct comparison between the uncertainty announced by the different software and the difference between the computed enrichment value and the value of the reference material. The methodology developed in order to evaluate the results of the experimental design is presented in a second part. It is a calibration process which consists in a first step of determining a statistics-based criterion to stop each acquisition once the result has converged, evaluating the dominating parameters of the experimental design in a second step and finally implementing calibration tables with the objective to confirm or to improve our current knowledge of uncertainties for these uranium enrichment measurements. The last chapter exhibits the entire process of calibration of the code MGAU based on the current approach.

## 2. Experimental Setup

### *2.1. Material for the experimental design*

To set up the experiment, standard trioxide uranium $U_3O_8$ sources have been achieved. The $U_3O_8$ powder is packaged in cylindrical containers made of resin. Enrichment range varies from depleted uranium (0.3%) up to highly enriched uranium (90%) and is certified by mass spectrometry (characteristics given in the table 1). The sources are also certified in uranium content and mass composition and sealed in conformity with the standards NFM61-02 (mechanics) and NFM61-003 (sealing). Three types of geometries, with the same resin bottom thickness of 2mm, have been designed in order to have a good compromise between the gamma emitting surface and the available mass for the manufacturing: the A type capsule (DxH: 38x5mm[1]) containing around 10g of uranium, the B type capsule (DxH: 50x5mm) containing between 12g and 33g of uranium and the C type capsule (DxH: 48x20mm => 50x33mm) containing between 86 and 120g of uranium.

| Source | Capsule type | U mass (g) | $^{234}U$ (%) | $^{235}U$ (%) | $^{236}U$ (%) | $^{238}U$ (%) |
|---|---|---|---|---|---|---|
| 8426-1 | B | 16.77 (0.32) | 0.003 (0.001) | 0.341 (0.01) | < 0.001 | 99.656 (0.012) |
| 8426-2 | B | 15.3 (0.32) | 0.004 (0.001) | 0.539 (0.01) | < 0.001 | 99.457 (0.012) |
| 8426-20 | C | 120.9 (0.36) | 0.006 (0.001) | 0.714 (0.01) | < 0.001 | 99.28 (0.012) |
| 8426-4 | B | 14.68 (0.36) | 0.011 (0.001) | 1.532 (0.01) | 0.002 (0.001) | 98.455 (0.012) |
| 8426-5 | C | 96.50 (0.30) | 0.023 (0.001) | 3.038 (0.01) | < 0.001 | 96.939 (0.012) |
| 8426-6 | B | 12.46 (0.32) | 0.035 (0.001) | 5.49 (0.01) | < 0.001 | 94.475 (0.012) |
| 8426-7 | B | 14.60 (0.32) | 0.079 (0.001) | 7.055 (0.01) | < 0.001 | 92.866 (0.012) |
| 8426-8 | B | 32.62 (0.36) | 0.076 (0.001) | 11.304 (0.036) | 0.021 (0.001) | 88.599 (0.012) |
| 8426-9 | B | 12.87 (0.36) | 0.105 (0.001) | 14.21 (0.036) | < 0.001 | 85.685 (0.012) |
| 8426-10 | A | 8.73 (0.5) | 0.147 (0.001) | 21.902 (0.036) | < 0.001 | 77.951 (0.012) |
| 8426-16[2] | B | 13.82 (0.32) | 0.0171 (0.001) | 26.075 (0.036) | 0.006 (0.001) | 73.748 (0.012) |
| 8426-11 | C | 87.45 (0.30) | 0.299 (0.001) | 29.187 (0.036) | 0.004 (0.001) | 70.51 (0.012) |
| 8426-12 | A | 8.5 (0.32) | 0.68 (0.001) | 45.112 (0.036) | < 0.001 | 54.208 (0.012) |
| 8426-13 | A | 9.88 (0.32) | 0.514 (0.001) | 57.042 (0.036) | 0.001 (0.001) | 42.443 (0.012) |
| 8426-14 | A | 12.34 (0.32) | 0.427 (0.001) | 68.043 (0.036) | 0.001 (0.001 | 31.529 (0.012) |
| 8426-15 | C | 86.04 (0.30) | 1.1 (0.001) | 89.303 (0.036) | 0.01 (0.001 | 9.588 (0.012) |

Table 1: certified characteristics of the sources

Three sets of $\gamma$ acquisition systems, containing one or more high purity germanium detectors and electronics but the same acquisition and $\gamma$ rays treatment software were implemented to realize the experimental design taking into account the three configurations routinely used in the field for uranium or plutonium isotopic composition measurements.

- The first set “UPu Inspector”, dedicated to the direct measurement of the isotopic composition of uranium with the code **MGAU** (version 2.2 provided by Canberra), consisted of a planar detector (“GL1015R” or “BE1015R” from Canberra), a multi channel analyser “Inspector” or “Inspector2000” from Canberra and the software “Genie 2k 2.0”.
- The second set “MGA++”, dedicated to the direct measurement of the isotopic composition of uranium using the code **U235view** (version 1.06 provided by Ortec), consisted of a planar detector (“GL0515R” or “GL1015R” or “CPL5” from Canberra), a multi channel analyser “DSPEC” or “DSPP” provided by Ortec and the software “Gammavision 6.01”.
- The third set “FRAM”, dedicated to the direct measurement of the isotopic composition of uranium using the code **PCFRAM** (version 4.2 provided by LANL, with the parameter set “U121_1001Coax.pst”), consisted of a coaxial detector “GEM25175” from Ortec, a multi channel analyser “DSPEC” provided by Ortec and the software “Gammavision 6.01”.

The spectra acquired in the three configurations are available for the evaluation of the code IGA, which is able to analyse any kind of spectrum — as the user only needs to adjust the energy in the first and the last channel and the FWHM and channel of a peak of interest. In the same way, the spectra acquired with the planar detectors are available for the evaluation of the code PCFRAM using the parameter sets “U100keVLEU” and “U100keVHEU” for the uranium analysis of the planar detector spectra employing the 100 keV region.

The previous $\gamma$ spectrometry systems are based on the acquisition of a $\gamma$ emission spectrum for which it is useful to follow the evolution of measured enrichment versus elapsed time so far, as since the result of measurement and its uncertainty tend towards a limiting value.

[1] D x H = internal Diameter x $U_3O_8$ filling Height; [2] Source $U_3O_8$ + $CaF_2$

Thus, the analysis of spectra acquired during a measurement by the codes MGA, MGA++, PCFRAM or IGA is coupled with the self-incrementing software “AutoISO_PLUM” [7], developed at IRSN in order to drive a self-incremented $\gamma$ spectrometry acquisition according to time, to visualize in real time on a graph the evolution of the measured quantity, which can be for instance the isotopic composition of uranium calculated by one of the 4 codes previously quoted, and to control indicators such as the counting statistics in the spectrum (cf. figure n°1). For this purpose, the user selects the number of spectra to be saved and analysed during a measurement, the measurement type, the code to be used, and the time interval between two successive acquisitions which can be cumulated or repeated. This tool helps the user to follow the convergence of a measurement, in order to stop it when there is no more significant improvement of the result and its precision with the counting time. All the relevant data accessible in the report issued by the code (live time, total counting in the spectrum, $^{234}$U, $^{235}$U and $^{238}$U contents and associated uncertainties for MGAU) calculated for each time step are stored in an Excel file that can easily be exploited.

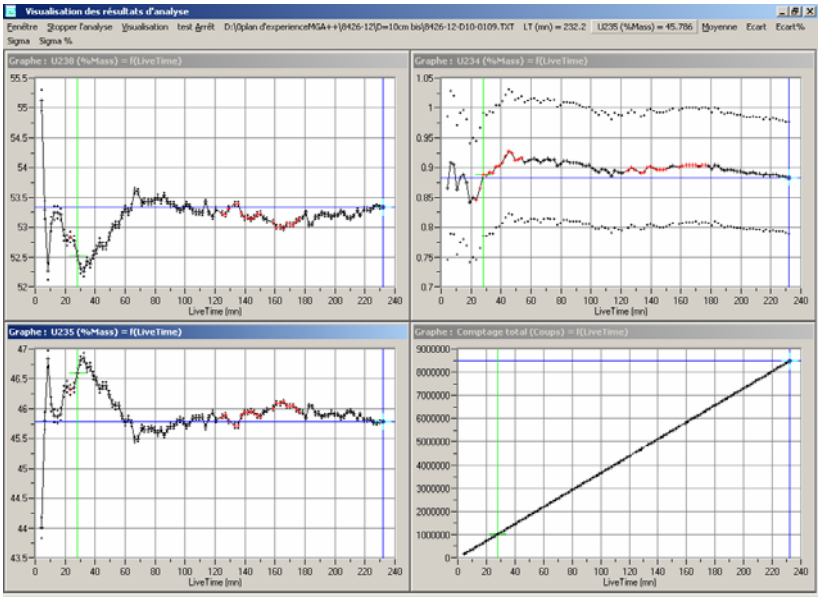


Figure n°1 : example of behaviour of the isotopic abundances of uranium ($^{235}$U, $^{235}$U, $^{238}$U) and the statistics in the spectrum according to the measurement time

The developer of the code provides a simple criterion to decide when to stop the acquisition, based on the total count in the spectrum with a recommended value greater than $10^6$, which is just a general indication. The measurements performed in the scope of the experimental design have been voluntarily acquired on a long time period above this criterion in order to study the influence of statistics on the results and to determine a reliable criterion for the convergence of the results.

### *2.2. Parameters for the experimental design*

The purpose of the experimental design is to highlight the factors which influence the result given by the software at the end of a measurement, and to determine their relative importance. The following parameters have been chosen in order to fulfil — as accurately as possible — realistic on-site measurement configurations:

- **$^{235}$U enrichment**: 16 $^{235}$U enrichment values, ranging between 0,35% and 89% (cf. table 1) ;
- **Source-detector distance**: Two distances are considered: 10 cm (chosen in order to keep measurements away from the sum peaks phenomenon; this distance may vary depending on the dead-time of the acquisition system) and 45 cm (distance at which one generally stands when one deals with waste drums);
- **Shielding**: interposition of a screen between the source and the detector in order to evaluate the influence of the container. Measurements are taken with 2 thickness stainless steel screens: 2mm, and 6mm, which correspond to conditions really met in the field;
- **Effect of the operator**: The operator effect is simulated by slightly varying the adjustment of the acquisition gain of the measuring equipment, i.e. by modifying the gain so that the peaks of stronger energy are shifted of more or less 3 channels compared to their normal position. In the case of MGA and

MGA++, the peak with 185,7 keV is shifted of more or less 3 channels (configurations 1 and 2). In the case of FRAM, the peak with 185,7 keV is shifted of 3 channels (configuration 1) and the peak with 1001 keV is shifted of 3 channels (configuration 2).

- **Matrix effect**: verifications of $^{235}U$ enrichment for safeguards purposes are usually done on uranium-bearing products ($UO_2$, $U_3O_8$,…) as well as waste drums containing uranium in suspension in vinyl or metal matrices. Therefore the influence of this parameter is studied using 200 litres "mock-up" drums available in the laboratory, containing vinyl matrices of density 0.2g/cm3 (average case generally encountered) and 0.4g/cm3 (penalizing case, also met on site), in the centre of which a source of given enrichment is put down.

The output parameters, measured and studied are the **uncertainty** given by the software and the **discrepancy** between the certified enrichment of the source (reference material) and the value given by the software.

### *2.3. Measurement configurations*

The experimental design represents an overall set of 224 measurements, 14 measurements being performed for each source (i.e. each enrichment) as stated in table 2 below. Depending on the measurement configuration, one measurement can be performed on a time period varying from 1 or several hours (for the basic configuration) up to several days (for the sources introduced in the mock-up drum), with a time step between each acquisition varying from 2 minutes up to 1 hour.

The configuration which involves a source alone placed at a distance of 10 cm from the front face of the detector is considered as the basic one.

| Source | screen (mm) | Distance (cm) | Matrix ($g/cm^3$) | Gain |
|---|---|---|---|---|
| 1 | 0 | 10 | 0 | 1 |
| 1 | 0 | 10 | 0 | 2 |
| 1 | 0 | 10 | 0 | 3 |
| 1 | 2 | 10 | 0 | 1 |
| 1 | 2 | 45 | 0 | 1 |
| 1 | 6 | 10 | 0 | 1 |
| 1 | 6 | 45 | 0 | 1 |
| 1 | 6 | 10 | 0 | 2 |
| 1 | 6 | 10 | 0 | 3 |
| 1 | 0 | 45 | 0 | 1 |
| 1 | drum thickness | 45 | 0,2 | 1 |
| 1 | drum thickness | 45 | 0,4 | 1 |
| 1 | drum thickness | 45 | 0,4 | 2 |
| 1 | drum thickness | 45 | 0,4 | 3 |

**Source 1 : $^{235}U/U_{total}$ = 0,35%**

**Source/detector distance** :
1. 10 cm
2. 45 cm

**Thickness of the screen** :
1. 2 mm
2. 6 mm

**Gain of the acquisition system** :
1. usual adjustment
2. 185 keV shifted of +3 channels
3. 1001 keV shifted of -3 channels
   (or 185 keV shifted of -3 channels)

**Matrix density** :
1. 0,2 $g/cm^3$
2. 0,4 $g/cm^3$

Table 2: acquisitions of the experimental design for each enrichment

Some repeatability measurements have also been performed in the basic configuration (source alone at a distance of usually 10 cm from the detector) in order to check the uncertainty given by the code according to its reproducibility.

## 3. Methodology of the evaluation

We propose hereafter to evaluate the performances of a code by performing a calibration which takes into account realistic in situ acquisition conditions as defined in the experimental design. The evaluation is executed in three successive steps. The first step consists in developing a statistics-based criterion to stop each acquisition at the beginning of the convergence according to the measurement time, in order to have an uncertainty which is representative of the result. The second step consists in analyzing the sensitivity of the measurement results (Comparison between the actual difference between the enrichment obtained with the code and the reference value and uncertainties also given by the code), compared to the variations of the 5 parameters of the experimental design. The third step consists in developing calibration tables which improve the enrichment values and the uncertainties given by the code.

### *3.1. Methodology used to develop the stopping criterion*

The objective here is to detect, for each measurement, the region where the acquisition starts to converge according to the $^{235}U$ enrichment value. For this purpose, a criterion named "relative standard deviation" has been developed, which consists of calculating the variation of the results ($^{235}U$ enrichment values) on N successive points of the data set which is analyzed. The relative standard deviation of this data set of N

points is, for example at the analysed point $P_i$ (where $\sigma()$ is the standard deviation and $A()$ is the average):

$$\frac{\sigma(P_i, P_{i+1}, ..., P_{i+N-1})}{A(P_i, P_{i+1}, ..., P_{i+N-1})}$$

This relative standard deviation is then compared to a threshold value S. If this relative standard deviation is less than the threshold value, then the first point of this data set is taken as the stopping value. The corresponding $^{235}U$ enrichment value will be recorded as the result of the measurement.

The values of parameters N and S are defined following a dichotomy procedure for optimizing them, starting with a number of points N = 15 (one chooses a high number in order to obtain the most restrictive possible criterion) and trying to seek the threshold value S starting with boundary values of 0.001 and 0.05.

### *3.2. Methodology used to evaluate the dominating parameters*

The knowledge of the parameters which mostly influence the measurements governs the implementation of the calibrations for the improvement of the results: one should act on the dominating parameters to correct the values provided by the codes. Two methods are applied, which should lead to the same conclusions: a statistical one and a probabilistic one.

**The statistical method** consists in assuming the existence of a linear relation between the input variables and the output variables. For each output variable ($Y$), we consider that:

$$Y = a_0 + \sum_i a_i X_i + \varepsilon$$

where $a_i$ is the coefficient of the linear approximation for the input variable $X_i$ and $\varepsilon$ is the error between real results ($Y$) and those obtained by linear approximation. The coefficients $a_i$ are found by using the least-square method to minimize the error represented by $\varepsilon$. The coefficients of the linear approximations thus contain information about the impact of each input data on the quantity studied: the higher this coefficient is, the stronger the influence of the parameter is. The coefficients are not compared just as they are: it is necessary to normalize them before, in order to take into account the range of the variations of the input parameter compared to those of the output parameter. The standardization corresponds to the following change (where $\sigma()$ is the standard deviation):

$$a_i^n = \frac{a_i \times \sigma(X_i)}{\sigma(Y)}$$

**The Probabilistic method** consists in calculating the variance of the results obtained in a specific configuration (for example all the measurements carried out without screen, then with a screen of 2 mm, then with a screen of 6 mm, if we consider the screen parameter). We obtain conditional variances which are divided by the variance calculated on the full result bank of the experimental design, which makes it possible to compare them. Thus, the higher the conditional variance, the more sensitive to the studied parameter the software is (it gives sometimes a weak variation, sometimes a strong variation).

The statistical method gives absolute information: one knows that that some parameter has an influence on the result, but one doesn't know if it is the case under all the conditions of measurement or some in particular. This method must be used at first in order to detect the dominating parameters. Then, the probabilistic method has the advantage of being more flexible than the statistical method: one can test all the configurations of interest (i.e. to implement classes according to the enrichment and the matrix, the enrichment and the screen, etc…). Moreover, we do not make any assumption here: we take account of the true values. That gives more precise results: one knows which parameters are influential and under which conditions. The two methods are complementary.

### *3.3. Methodology used to improve the results (enrichments and uncertainties)*

The implementation of calibration tables gives an interesting overview of the performances of a code depending on the measurement configuration, by considering both $^{235}U$ enrichment value obtained and the uncertainty provided by the code.

The calibration tables of the $^{235}U$ enrichment measured values are developed by gathering measurements in homogeneous categories (i.e by distinguishing from the others the measurement conditions defined by the dominating parameters). The average of the difference between the measured $^{235}U$ enrichment value and the reference value for each measurement is calculated inside each category. This mean value represents

the correction to be applied to current measurements under the same conditions, which corresponds in more a general way to the observed bias.

The calibration tables of the uncertainties are developed by gathering the measurements in the same way. The average of the difference between the relative uncertainties given by the code and the relative errors actually made by the code on the true value determined for each measurement is calculated inside each category. This mean value represents the correction to be applied to the relative uncertainties currently provided by the code under the same conditions, which is a signature of the representativity of the announced uncertainties.

### *3.4. Methodology used to evaluate the influence of the counting statistics on the result*

The stopping criterion defined above is set up on the basis of the evolution of the result according the measurement time. Its consistency is checked by reprocessing the same statistical analysis for the time T/2 and 2T with T being the stopping criterion time. However in practice, measurement times of more than 30 to 60 minutes are rarely allocated for safeguards verifications.

The analysis of the connections between the statistics in one or several peaks of interest and the measurement results ($^{235}$U enrichment and uncertainty) constitutes another possible track for the construction of an objective and physics-based stopping criterion. The point consists in determining if the convergence zone corresponds to similar counting statistics for the peaks studied by measurement category, then potentially developing a law which makes the evaluation of the expected precision possible in a given measurement configuration, for weaker statistics.

## 4. Results obtained with the code MGAU

The following paragraphs exhibit the results of the evaluation and calibration process of the code MGAU, applied to the 153 measurements of the experimental design performed with the "UPu Inspector" $\gamma$ spectrometry system. Only 153 measurements were taken (whereas 224 where planned), as of the first tests it appeared that the parameter gain did not have impact on the results. Measurements to 45 cm with a screen of 2 mm were not carried out; the impact of the screen is appreciated through other measurements.

### *4.1. Stopping criterion*

The parameters N (number of cumulated acquisitions) and S (threshold) implemented for the application of the stopping criterion to the "UPu Inspector" $\gamma$ spectrometry system are given in table 3. It was not possible to develop a single criterion for all of the measurements: the behaviour of the measurement system is different in the presence of a screen or of a matrix, and according to the $^{235}$U enrichment of the source.

| no screen<br>no matrix | with screen<br>no matrix | with screen<br>with matrix |
|---|---|---|
| $\%^{235}U < 1\ \%$<br>N = 14; S = 0.008 | | |
| $\%^{235}U > 1\ \%$<br>N = 14; S = 0.001 | $\%^{235}U > 1\ \%$<br>N = 14; S = 0.0025 | $\%^{235}U > 1\ \%$<br>N = 14; S = 0.0018 |

*Table 3: parameters of applicability of the Relative Standard Deviation criterion*

Once the criterion is satisfied, the acquisition converges towards a $^{235}$U enrichment value. Under these experimental conditions, we will be able to qualify the uncertainties.

### *4.2. Repeatability measurements in the basic configuration*

The acquisitions were performed for each of the 16 sources in the basic configuration with counting statistics taken in the convergence region of each basic measurement. The synthesis is presented in table 4.

The enrichment value given by MGA is stable from one measurement to another whatever the $^{235}$U enrichment is (variance < 1%). It is the same for the announced uncertainties. The relative differences between measured and certified enrichment vary between -8% min and 15% max for the depleted or natural uranium and between -4% min and +5% max for enriched uranium. The uncertainty given by the code seems to cover the real error committed by the code in the majority of cases, considering a value of around 4-8% for depleted and natural uranium and a value of around 1% for enriched uranium (cf. table 4). As stated in ref. [5], our measurements shows a tendency to overestimate the enrichment for depleted uranium, but the uncertainties are quite high in our study (cf. "%$\Delta$>0" column in table 4). Most of the measurements done with high enriched uranium (>20%) underestimate the true value, but it is not systematic, perhaps it is due to the measurement precision which is only around 1-2%.

### *4.3. Evaluation of the performances of the code (uncertainty components)*

The dominating parameters for the relative differences between measured and certified enrichment are the enrichment and the matrix. According to the boundaries chosen for the screen thickness, its influence is less important. The figures 1 to 3 exhibit the difference between the measured enrichment and the certified value, for the optimal configuration with the best precision (figure 2), for a configuration with a stainless steel screen put between the sample and the detector (figure 3), and for a configuration with a mock-up drum (figure 4).

| % $^{235}$U | Nb of Acquisitions | Average value of abs(Δ) | Average value of Unc. | % (Unc > Δ) | % (Δ>0) |
|---|---|---|---|---|---|
| 0.341 | 86 | 5.8 | 8.7 | 80% | 88% |
| 0.539 | 32 | 6.0 | 4.5 | 31% | 100% |
| 0.714 | 76 | 4.0 | 7.0 | 86% | 63% |
| 1.532 | 42 | 2.7 | 1.3 | 14% | 100% |
| 3.038 | 31 | 0.6 | 0.9 | 77% | 48% |
| 5.49 | 52 | 1.7 | 0.9 | 13% | 98% |
| 7.055 | 289 | 0.9 | 1.1 | 66% | 68% |
| 11.304 | 12 | 0.6 | 0.7 | 67% | 8% |
| 14.21 | 176 | 0.8 | 0.9 | 65% | 25% |
| 21.902 | 36 | 0.4 | 0.6 | 81% | 67% |
| 26.075 | 224 | 1.0 | 1.2 | 68% | 34% |
| 45.112 | 21 | 1.3 | 1.6 | 62% | 43% |
| 57.042 | 97 | 0.6 | 0.8 | 67% | 43% |
| 68.043 | 191 | 1.0 | 1.1 | 63% | 28% |
| 89.303 | 104 | 0.8 | 1.7 | 95% | 81% |

Δ = relative discrepancy between the measured $^{235}$U enrichment and the reference value (%); Unc = code relative uncertainty (%)

Table 4: repeatability measurements synthesis

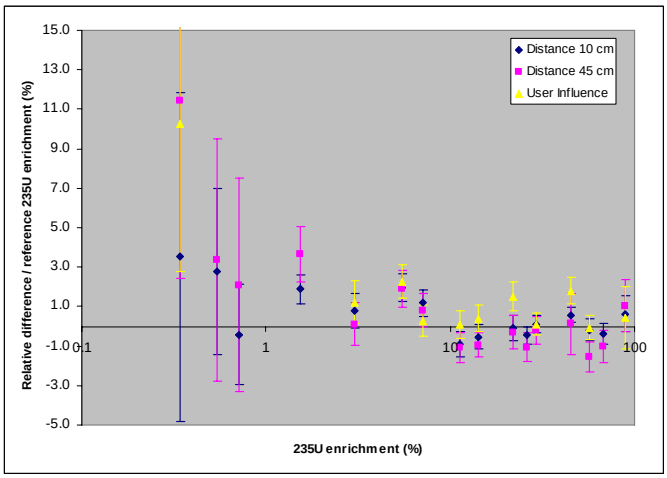


Figure 1: optimal configuration, distance and user effect

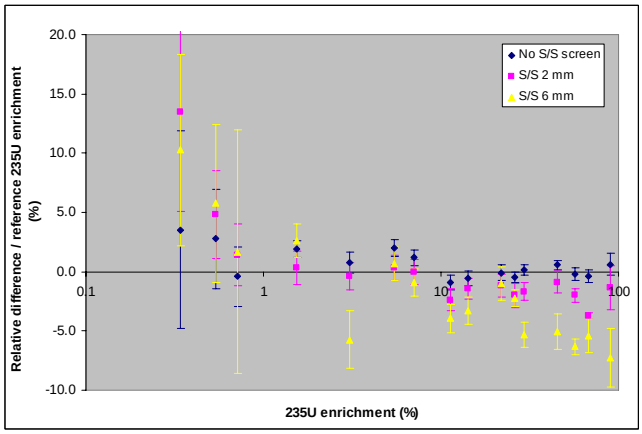


Figure 2: optimal configuration, stainless steel screen effect

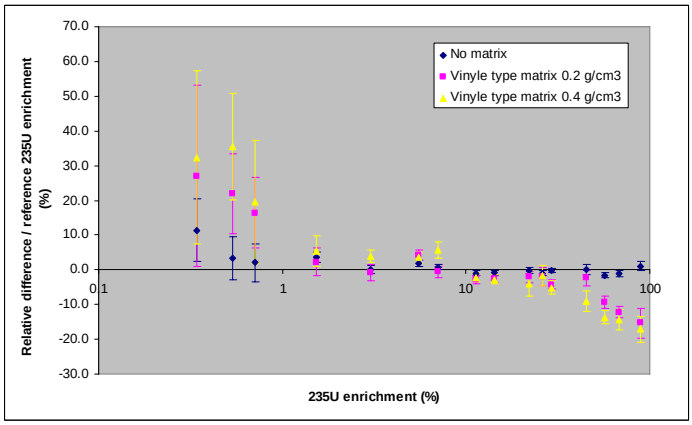


Figure 3: 45 cm configuration, waste drum effect

Regarding $^{235}$U enrichment in the optimal configuration, as expected the best accuracy of less than 1% is obtained in the middle of the range around 10%, whereas depleted uranium is the most difficult to evaluate (around 3%). The measurement uncertainties are weaker at the extremes, even if it is much less pronounced for high enriched uranium. The presence of a screen between the source and the detector has a limited influence on the $^{235}$U enrichment, which is in agreement with the measurements reported in ref. [5]. The screen effect is observed with a stainless steel absorber of thickness 6 mm, for high enriched uranium > 30%, which causes a discrepancy of around 3-4% compared to the optimal configuration. The matrix is the most influent parameter, which results in discrepancies of more than 15% at the extremes of the range.
The distance and the operator adjustment of the gain do not have any influence on the results, considering the boundary values chosen for our study (figure 2). The tendencies of overestimation of depleted and natural uranium and underestimation of high enriched uranium already pointed in reference [5] seem to be confirmed by the shapes of the figures especially when a screen or a matrix is interposed between the source and the detector.

The application of the calibration methodology described in the third chapter, implemented with the results issued from the application of the stopping criterion previously defined, leads to the values exhibited in the tables 4 (enrichment corrections) and 5 (uncertainty corrections), which distinguish 4 enrichment classes (depleted and natural, low enriched uranium up to 10%, enriched uranium between 10 and 50%, and high

enriched uranium from 50% up to 90%) and 3 separating measurement configurations: the optimal one without any absorber, the configuration with a stainless steel absorber and the configuration of a waste drum with different matrixes. The values presented in table 4 have been converted to relative values compared to the median value of the enrichment in each class considered, for representativity reasons.

| measurement conditions | 0 to 1% | 1% to 10% | 10% to 50% | 50% to 90% |
|---|---|---|---|---|
| no screen no matrix | 4,4% | **0,6%** | **0,2%** | **0,0%** |
| with screen no matrix | 2,0% | **-1,1%** | **-2,6%** | **-3,8%** |
| with screen with matrix | 25,4% | **1,9%** | **-3,1%** | **-14,4%** |

*Table 4 : calibration table for the measured $^{235}U$ enrichments*

| measurement conditions | 0 to 1% | 1% to 10% | 10% to 50% | 50% to 90% |
|---|---|---|---|---|
| no screen no matrix | 3,84% | -0,26% | 0,16% | 0.92% |
| with screen no matrix | 13.66% | -0,12% | -1.21% | -0.40% |
| with screen with matrix | -1.81% | -0,21% | -0.87% | -10.14% |

*Table 5 : calibration table for the uncertainties*

The values given in table 4 should represent the correction to apply to measured enrichments, after convergence, in order to improve the result compared to the certified value. Only the bold values are reproducible with a measurement time multiplied by two with reference to the time given by the stopping criterion, which means that the criterion needs to be improved for depleted and natural uranium. The corrections to apply to the relative uncertainties given by the code at the convergence time, calculated in table 5 using the results issued from the stopping criterion give an idea of the representativity of those uncertainties at a given moment (underestimation for negative values), but they are probably not representative in all cases. The definition of a stopping criterion using statistics concepts only, based on the shape of the behaviour of the measured value according to time, does not seem to be the most appropriate approach to evaluate the expected uncertainties. The study is currently moving towards the evaluation of the evolution of a result according to the statistics and/or the shape of the peaks of interest used by the code to analyse a spectrum, that is to say the definition of a more physics-related stopping criterion.

## 5. Conclusions

The paper reports on an experimental design that has been performed on $U_3O_8$ standard for the determination of $^{235}U$ enrichment. The experimental design represents an overall set of 224 measurements, 14 mesurements being performed for each $U_3O_8$ standard taking into account the five following influencing factor: $^{235}U$ enrichment, source-detector distance, shielding, effect of the operator, matrix effect. Three sets of $\gamma$ acquisition systems have been implemented.

The main goal of such an experimental design is the mastery of the uncertainties and a good knowledge of the fields of applicability of the isotopic composition determination codes for a better use on site. The originality of this experimental work lies in the approach used to evaluate the performances in several realistic in situ acquisition configurations, based on incremented acquisitions according to measurement time, and a calibration process to treat the experimental results and assess the performances that can be reached under several statistical conditions.

The first results of the statistical analysis, which is reported in this paper, have been applied to the experimental data obtained with the code MGAU with the objective (1) to develop and validate the methodology, and (2) to demonstrate it on this code. The results issued from the code MGAU confirm some of the tendencies already pointed in reference [5] and highlight the limitations of the code for waste drum assay. The evaluation of an adequate stopping criterion for an incremented measurement needs to be further studied. Once the analysis method is validated, it will be used for the other codes.

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# Session 13

# IS-Implementation I

# How much Safeguards is Needed in Europe?

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**Abstract**

It is expected for AD 2008 that Integrated Safeguards (IS) will be implemented in the European Union (EU) as a whole or, at least, in most of its Member States. Furthermore, it is expected that, by this time, the EU and the International Atomic Energy Agency (IAEA) will have settled a new partnership approach with a new efficient and effective control regime taking into account the special situation in Europe. The application of the state-level approach in connection with the implementation of the Additional Protocol (AP) will offer the IAEA a powerful tool to reach this goal. The paper discusses, on the basis of the German experiences, the potential of the state-level approach in terms of distribution of inspection effort. If appropriate credit is given to the strengthened institutional and economic European structures, the on-site inspection effort could be reduced. The evolution of Europe to a common political and economic entity offers, besides other aspects, a huge range of additional information which is available for the IAEA and is of extreme value for safeguards implementation and evaluation. A possible reduction of routine inspection effort in the field will be overcompensated in quality by the multiplicity of safeguards relevant information under Integrated Safeguards. As in the past, European countries like Germany will continue to be countries where a high quality level of international safeguards is given by the international interconnections of the European fuel cycle. The structure of this fuel cycle will also in the future render any clandestine nuclear activities of individual Member States practically impossible. Therefore, the focus will change from routine on-site inspection to analyzing relevant information about institutional and economic structures, processes, and communication within the EU that will be available and even grow in the future.

**Keywords:** Integrated Safeguards, state-level approach, information analysis, inspection effort

## Introduction

In the last decade, there has been a development of greatly enlarging the scope of international safeguards. On the one hand, the scope of investigations increased, i.e., with regard to detecting clandestine nuclear activities, and, on the other hand, the variety of safeguards measures available to the Agency increased, too. The safeguards strengthening measures triggered by the Additional Protocol and endorsed by the International Atomic Energy Agency's (IAEA) Board of Governors, have given the Agency the necessary legal authority to widen the scope of its activities, now ranging from the verification of declared nuclear material inventories and declared inventory changes to the investigation of all nuclear and nuclear related activities in a state. For states with comprehensive safeguards agreements (i.e., INFCIRC/153-type agreements), the Agency is expected not only to verify that declared nuclear material is adequately accounted for but also to provide assurance about the absence of undeclared nuclear material and nuclear activities in a state.

Although it is evident that it is not possible to indisputably prove that undeclared activities do not exist in a state, the Agency can provide a sound assurance that there is no indication for such activities. The basis for such a statement about the absence of undeclared nuclear material or nuclear activities in a state is a comprehensive analysis of all relevant information available to the Agency, the IAEA's increased access to nearly any location in a state and the other technical measures the Additional Protocol provides for.

The more appropriate information about a state's behaviour in the nuclear field is available and evaluated to confirm the compliance of the state with the safeguards obligations, the more assurance can be given to the safeguards conclusion.

## Different Information Sources

As a result of the safeguards strengthening measures and, in particular, of the provisions of the Additional Protocol, the Agency has at its disposal a substantially increased amount of information about a state's nuclear programme. This should allow the Agency to establish a comprehensive view about the scope and nature of a state's nuclear fuel cycle and related activities.

This information includes:

- information provided by a state according to its obligation under the safeguards agreement, e.g., nuclear material accounting reports and design information;
- information provided according to Article 2 of the Additional Protocol, e.g., information about: R&D in the nuclear field, nuclear sites, industrial capabilities in the nuclear area, exports of specified items, long term plans in the nuclear field;
- information obtained by the Agency through its verification activities conducted under the safeguards agreement and Additional Protocol, e.g., the results of inspection activities, design information verification activities, complementary access activities; analysis of environmental samples;
- information made available to the Agency through co-operation and voluntary measures of states, like Member States Support Programme activities or participation in the voluntary reporting scheme;
- information available to the Agency through its activities in non-safeguards areas like technical co-operation, nuclear safety, nuclear databases;
- information obtained from satellite imagery analysis;
- open source information, e.g., information issued by commercial companies, universities, different types of public organisations, governments, administrations.

As can be seen from this list, the state continues to be a very important source of information. States that signed the Additional Protocol have made a political commitment to provide transparency in their nuclear programmes and, thus, to enable the Agency to develop an extensive view of the state's behaviour in the nuclear field. This is accomplished through the provision of information as required by the Additional Protocol declarations and through co-operation and a broad information exchange on many different levels. In the case of the European Union states, which are also subject to European Commission safeguards, there have been, since many years, permanent activities that help to increase the transparency and mutual understanding, e.g., by common inspection planning, sharing of inspection equipment, and sharing of inspection results. In addition, there are regular meetings between IAEA, EU Commission and national authorities that should allow the Agency to build up a far reaching understanding of the behaviour in the nuclear area. In Germany, the Agency was able to eyewitness the development in the society, government, science and industry from nuclear enthusiasm in the nineteen sixties and seventies to growing scepticism in the nineteen eighties and nineties to the still valid governmental decision taken in the year 2000 to phase out the use of nuclear energy for electricity production.

Another very important source of information are activities of the Agency itself. Besides the information collected through its inspection activities in safeguards, i.e., verification and Complementary Access (CA), the IAEA has at its disposal a broad range of other nuclear related information. It emerges from the Agency's technical co-operation programmes, research and development activities, and information services offered to the public, i.e., INIS database and IAEA's activities in the field of nuclear safety and security. Last but not least, the IAEA personnel of more than 2 thousand multi-disciplinary professionals and support staff members from more than 90 countries represents an almost inexhaustible human knowledge base for the interpretation and understanding of cultural and national aspects of conduct of states in the nuclear field.

The third information source, namely open source information, is a nearly unlimited resource in the European Union. As an example, the freedom of information legislation grants everybody access to information held by all kinds of government bodies; academic rivalry and competition for public funding spawn floods of publications in scientific areas; publicly held corporations are subject to strong rules concerning their disclosure obligation to publish relevant information. As a general fact, this prolific nature makes it also most difficult to utilize open source information. Commercially published scientific literature is still subject to well-defined channels or systems of publication and distribution and to

bibliographic control. As soon as non-conventional literature or grey literature is concerned, these mechanisms do dot necessarily apply, especially with regard to media reporting and the Internet. Another aspect is that open source information, especially media and Internet, are the most actual sources of information.

## Effort and Ways to Analyse Available Information

To make proper use of all this different kind of information it must be collected and evaluated. Depending on the type of information, this task may require different amounts of effort.

The Agency is applying information analysis since the start of its safeguards activities. Design information and accountancy information are an integral part of safeguards since the beginning. The Agency since long has developed the necessary capabilities to analyse and verify this type of information that can be characterised as well-structured information, assembled and delivered in a predefined format. Information delivered by the state according to the requirements of the Additional Protocol (AP) is of a similar nature. The guidelines for the preparation and submission of AP declarations describe in detail the format how to compile this information.

Information collected or generated by the IAEA in non-safeguards areas is gathered for different purposes and needs to be assessed with regard to the contribution it can deliver for safeguards purposes. The advantage here is that the IAEA is familiar with the structure, knows about the reliability of the information and can profit from synergies if the same information is analysed with regard to different purposes.

The most challenging task concerning information analysis is posed by open source information. This information must be searched, collected and evaluated. Since much of it is available in electronic form, the search and collection can be supported by sophisticated software tools and even be conducted in a semi-automated way. Evaluation requires far more human intervention, starting with the task to properly understand the significance of the information, to judge the relevance for safeguards, to check the reliability of the information's origin, to put the information in the context of other available knowledge and information, and so forth. This evaluation task requires particular skills and expertise in information handling, knowledge about the environment within the information is generated, in order to make a sound judgement on the reliability of the source and the intention behind of producing the publication, and, last not least, technical knowledge in the areas of the nuclear fuel cycle

In recent years, the Agency has established substantial additional information review and evaluation capabilities. A central element in these efforts is the state evaluation process, a continuous process of evaluating all information available to the Agency about a state's nuclear programme and related activities for the purpose of planning safeguards activities in the state and of drawing safeguards conclusions about the state.

The level of confidence in judgements derived from the analysis of open source information is not just a question of the amount of information available but more a question of the structure and accuracy of the analytical process. In our view, the information analysis should not be based on a "mosaic theory", i.e., to collect as many bits of information as possible assuming that, when put together like a mosaic or puzzle, they may enable the analyst to gain a clear picture of reality. While the aim is to make a judgement on clandestine activities that in most cases do not exist, one can never be sure to have available all the information needed to complete the puzzle. This will result in a concentration of effort in the area of information collection and may cut down resources needed for the analysis.

In our opinion, a more appropriate model for the information analysis is the evaluation of hypotheses. If the IAEA becomes aware of an inconsistency between the declarations of the state and its own observations, it should, based on its specialised knowledge, develop hypotheses to explain this inconsistency and purposefully collect information to evaluate alternative hypotheses to the end of plausibility checks. This type of a more conceptually driven analysis should be the approach for a state level evaluation.

One rationale behind all these information evaluation efforts is the idea of Integrated Safeguards (IS). The overall objective of IS is to achieve "the optimum combination of all safeguards measures available to the Agency under comprehensive agreements and Additional Protocols which achieves

the maximum effectiveness and efficiency within available resources …". If a carefully conducted state evaluation does not show any indication for undeclared nuclear activities in the state, some proliferation scenarios like undeclared enrichment or undeclared reprocessing become less plausible, and this allows reducing on-site inspection effort while maintaining or even increasing safeguards efficiency. In such a case, the reduction of inspection effort in the field does not at all mean a reduction in safeguards coverage, but a new optimum composition of safeguards measures where information analysis effort substitutes for on-site verification effort.

## Information Available in EU States

In the European Union (EU), a huge amount of information is publicly available from many different and independent sources. In many areas, the disclosure of information is mandatory for the administration, for all sorts of institutions and also for private enterprises. The variety of information and the diversity of information sources very much facilitate a directed information collection to evaluate concrete hypotheses. The same variety and diversity will lead to an infinite effort required when the aim is to analyse the information according to the "mosaic" approach.

It is hard to imagine that a "mosaic" approach will lead to findings by an external analyst that are not already discovered and addressed before by an internal actor. The EU is a unique entity of now 27 states. This entity is based on a subtly balanced network of competences in all socially, politically and economically relevant areas with strong supranational elements and close intergovernmental co-operation. In the economic area, the basis is a common market without internal frontiers where a supranational body, the EU Commission, enforces competition in all main supply areas, including the energy market. On the basis of EU directives, state controlled sectors are being privatized. The EU Commission has the power to take states that fail to transpose the directives, to the European Court of Justice, and executes these rights. European Community competition law is a prominent area of strong authority of the European Union. This authority includes detailed control and close monitoring of direct and indirect aid given by EU Member States to companies.

EU membership cuts national sovereignty in many other aspects. For instance, national finances and deficit spending are closely monitored by the EU Commission. In fact, EU legislation covers a range as broad as national legislation of the member states. Where a conflict arises between EU law and national law of a member state, EU law takes precedence, so that the law of a member state must be amended.

EU States are embedded in many common undertakings as for example the Common Foreign and Security Policy (CFSP) that can be regarded as another step towards the transfer of national authority to a common foreign policy and even a common defence policy of the European Union. The European Council defines the principles and general guidelines for the CFSP as well as common strategies to be implemented by the EU, including capacities and structures for military crisis management, civilian crisis management, and conflict prevention and ensures that the Member States refrain from any action which is contrary to the interests of the Union.

Another policy area of common undertakings is research and development. The multiennial Framework Programmes for Research and Technological Development (FP) are the main instrument through which the EU implements its research policy. The EU intends to support and encourage research in key areas and to promote international research collaboration while offering funding to all kinds of public and private entities. All notable research institutions in the EU participate in the Framework Programmes and are thereby closely integrated into international scientific collaboration.

## Possible Consequences and Conclusions

There are several consequences to be considered. First, there is a huge amount of information easily accessible covering all aspects of the political and economical activities generated by many different actors in the European Union, which allows a directed search for and manifold comparisons of independently generated information. There is sufficient information available to the IAEA with regard to amount, quality and reliability to test any conceivable hypothesis of non-compliance with safeguards obligations of any desired Member State of the EU.

Concerning the implementation of Integrated Safeguards, this situation offers a broad scope of possibilities to enhance the efficiency and to define an optimum combination of safeguards measures and to replace on-site verification activities by information analysis.

If, as a second aspect, the view is broadened from the pure availability of the information to the context in which the information is generated, it becomes clear that any non-compliance of an EU Member State with safeguards obligations would be uncovered and made visible by the internal mechanisms of the European Union. The environment in which a state can act in the EU consists of a complex and subtly balanced machinery of distributed competences and networked co-operations in all areas. The contemplation and implementation of a nuclear proliferation policy by one or several EU Member States would inevitably create severe disturbances within this environment without any chance to remain undetected.

These principles do not only apply with regard to supranational or intergovernmental aspects but also within states themselves For instance, in Germany there are 4 different levels of independent and, quite often, even competing governmental administrations:

- EU level
- national level (Bund)
- subnational state level (Länder)
- local administration level

If we consider the nuclear domain, all these administration levels are involved in the processes of licensing and monitoring of activities and, thus, function as inherent barriers against nuclear proliferation.

The unique structure of the EU offers an additional huge potential for savings to the Agency. As long as the collusion of now 27 member states, including 2 nuclear weapons states, to cover the diversion of one or several members is not regarded as a plausible scenario, results of information analysis of supranational EU bodies could, at least in parts, be taken as a trustworthy assessment result. In this context, we mention the activities and assessments of the European Commission and other supranational bodies like European Court of Auditors, the European Court of Justice or the European Central Bank that monitor different facets of Member States' behaviour. To give one example, Article 35 of the Euratom Treaty requires that each Member State shall establish facilities necessary to carry out continuous monitoring of the levels of radioactivity in air, water and soil and to ensure compliance with the basic safety standards. This Article also gives the European Commission the right of access to such facilities in order that it may verify their operation and efficiency. This right is executed by the Commission services and results are published.

By far the greatest potential for saving could be exploited, if the Agency would accept the EU safeguards services as a source of reliable and unbiased information. The Agency knows very well the concepts, procedures and instruments applied by Commission safeguards services, since they are almost identical with the corresponding IAEA activities as far as the verification of declared material and declared facility layout are concerned and are even carried out jointly in the EU states under the INFCIRC/193 safeguards agreement.

Although the objectives and the scope of their respective safeguards activities as a whole are not identical, essential parts of their safeguards conclusions read quite similar and suggest the assumption that the IAEA could make more extensive use of assessments already made by the Commission for those parts of the safeguards conclusions.

The Commission's statement reads:

> *"… As a result of the verification activities undertaken by Euratom Safeguards in the framework of Chapter 7 of the Euratom Treaty, no evidence was found to suggest that nuclear materials were diverted from their intended uses. Nor was any evidence found to suggest non-compliance with particular safeguards provisions assumed by the Community under agreements concluded with non-EU States."*
> *[Communication from the Commission to the European Parliament and the Council; Euratom Safety and Security - Activities in 2003; COM(2004) 861 final]*

The IAEA's statement reads:

> *"… For those States, the Agency found no indication of the diversion of nuclear material placed under safeguards. On this basis, the Agency concluded that for these States, the nuclear material placed under safeguards remained in peaceful nuclear activities or was otherwise adequately accounted for."*
> *[IAEA Safeguards Statement for 2003; http://www.iaea.org/OurWork/SV/Safeguards/es2003.html]*

This does not mean that the right of the IAEA to derive its own independent conclusion will be restricted in any way. It is up to the Agency to determine to what extent and in what cases it may revert to information and assessments made by the Commission's safeguards services. However, under the current international situation, where the demand for the IAEA safeguards is on the increase, it is becoming increasingly important for the IAEA to improve the efficiency of its safeguards activities through the effective use of its limited safeguards resources. From such a viewpoint it might be beneficial to consider, if and when appropriate, in how far assessments of other supranational bodies could be taken into account to derive the safeguards conclusion.

A third aspect, also resulting from the specific situation in the EU, is the limitation of the investigation in the state level approach of the IAEA to individual national states. The Member States of the EU constitute a single common market without internal frontiers for the exchange of goods and services. Sensitive facilities in the nuclear fuel cycle, like enrichment services, are operated as multi-national companies. Formerly national companies merged and are now in public or multi-national ownership and operate in several or all countries of the EU. All EU Member States are subject to the provisions of the Euratom treaty, which means that all nuclear material is owned by the EU, is supplied only by the Euratom Supply Agency and is subjected to the Commission's safeguards services to verify the use of the material. In this situation it is, for example, not clear how to attribute the capabilities of the nuclear industry in the EU to individual states in a state level evaluation. On the other hand, it would make sense to pool similar facilities across national borders, e.g. power plants owned by the same company, to reach better savings when statistical sampling is applied in an Integrated Safeguards scheme.

Summarizing, we can state that the unique nature of the European Union calls for the consideration of the particular situation and of possible synergies stemming from the activities of other supranational institutions in IAEA's safeguards approaches for the Member States of the EU offering a large potential of savings to the IAEA while maintaining or even increasing the safeguards coverage in the EU.

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# The dialogue between the EU Member States and the EC services on the implementation of Euratom Treaty Safeguards

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***Abstract:***

On 27 February 2007 the Commission approved a Working Document SEC(2007)293 – entitled "Implementing Euratom Treaty Safeguards" (IETS) [1].
The Working Document combines the Commission's strategy for improved nuclear safeguards effectiveness and efficiency in the Community and the WPAQ's proposals on a new framework for Euratom safeguards. It records the large consensus reached between the Commission and the Council on their shared understandings about nuclear safeguards in Europe.
This paper reviews the process of dialogue between the Commission and the Council that led up to the consensus, highlighting some of the key features in the Working Document.

**Keywords:** Euratom, IAEA, safeguards, implementation, inspections, verification activities, installation types

## Introduction

On 30 April 2004, the Commission adopted a Communication [3] outlining the fundamental principles of a revised, improved, approach for meeting the objectives of the Community nuclear safeguards policy. A period of consultation with the Community stakeholders ensued resulting in the drafting of an agreed implementation paper[2].

The European Atomic Energy Community ("Euratom") was established in 1957 with a clear aim for the European Atomic Energy Community policies: a higher standard of living, stronger external relations and the speedy establishment and growth of nuclear industries.

The Treaty provided the Community with specific objectives to attain this aim structured along eight axes for action: research, safety and radiological protection, investments, supply, safeguards, nuclear material ownership, internal market and external relations. Amongst these, three are particularly important for the control of nuclear materials in the Community: supply (chapter VI) , safeguards (chapter VII) and external relations (chapter X). In this respect, the Treaty requires the Community to:

– ensure that all users in the Community receive a regular and equitable supply of ores and nuclear fuels making sure no discrimination occurs between users on grounds of the use which they intend to make of nuclear supplies, and guaranteeing that such use is not unlawful or found to be contrary to the conditions imposed by suppliers outside the Community;
– make certain, by appropriate supervision, that nuclear materials are not diverted to purposes other than those for which they are intended (as declared);
– establish with other countries and international organizations such relations as will foster progress in the peaceful uses of nuclear energy, including assuming safeguards obligations in agreement with a third State or an international organisation;

For meeting these objectives, the Euratom Treaty outlines the architecture of a nuclear material safeguards system in the Community. This "Euratom safeguards system" was designed to control in the broader sense the security of nuclear materials in the territories of the Community. It responds to the requirement of ensuring that nuclear materials are not

diverted for purposes other than those for which they are intended.

The Treaty entrusts the European Commission with the responsibility of administering the Euratom safeguards system on behalf of the Community. It explicitly sets forth in its Article 77, the two objectives for the Commission in implementing the system:

– Article 77(a) calls upon the Commission to satisfy itself that in the territories of Member States nuclear materials are not diverted from their intended uses as declared by the users;
– Article 77(b) calls upon the Commission to satisfy itself that the provisions relating to supply and any particular safeguarding obligations assumed by the Community under an agreement concluded with a third State or an international organisation are complied with.

## The Council Working Party on Atomic Questions

Within the framework of the discussions – held mostly during 2005 – between the Commission services and the Member States' representatives at the level of the Council, the Working Party on Atomic Questions (WPAQ), under UK presidency, drafted and agreed on a 'non-paper' [10] on a "New Framework for Euratom safeguards".

The WPAQ document elaborated a set of principles for Euratom Treaty safeguards, which the party members agreed would contribute to establishing a modern, effective and efficient means to satisfy both the requirements of Article 77 a) of the Euratom Treaty, and the Euratom Treaty safeguards system's role – as Community system of accounting for and control of nuclear material within international safeguards– under Article 77 b) of the Euratom Treaty. On 20 December 2005, the Council's "Comité des Représentats Permanents" (COREPER) took note of the WPAQ document.

The WPAQ considered then that modalities should be defined to enable discussion and agreement between the Commission and Member States on Euratom safeguards issues that require a common EU position, and in particular on the new safeguards approaches, while respecting the sharing of competencies between the Commission and the Member States.

At the same time, the WPAQ promoted the idea that a discussion on co-ordination with IAEA safeguards could take place in parallel to the updating on the revised approaches documents. Once the initial results of these updates and discussions would be available, Member States' experts would be called to examine them. This should take the form of WPAQ meetings in the composition of safeguards experts – e.g. a panel of national experts on nuclear safeguards –, as had been done for discussions of Commission Regulation 302/2005 [11]. The panel would be supported by an advisory group under Commission chairmanship. The safeguards experts would meet when required in order to ensure adequate follow up and review of progress of an indicative work plan:

– Further develop the revised approach and framework (defining NMAC auditing guidelines, new practices; refine the framework per type of installation; establish principles for randomly scheduled inspections)
– Discuss and agree cooperation arrangements with the IAEA (completing the transition of certain Member States to the Community/IAEA verification agreement; agreeing on updated partnership arrangements; monitoring the implementation of the Additional Protocol; achieving full transition to Integrated Safeguards)
– Review the implementation of the Commission Regulation No 302/2005 and in particular the Commission Recommendation 2006/40/Euratom [12]
– Examine the need for a regulation or guidelines concerning verification and inspection practices
– Monitor the evolution of international safeguards and its impact on the Community positions

## Building shared understandings

From 2005 to 2007, the WPAQ experts' panel and Commission advisory group met at several occasions: e.g. 29 May 2006 in Luxembourg, 5 October 2006 in Helsinki, 28 November 2006 in Luxembourg, 14 February 2007 in Bonn.

In preparation of the meetings, the Commission services took into account the WPAQ experts' comments for updating its approaches. The items raised by the WPAQ experts were discussed thoroughly during the meetings. The Commission had then the opportunity of clarifying its interpretation of all the key points above. This process eventually led to the

formulation of shared understandings between the Commission advisory group and the WPAQ experts about the key topics in discussion.

The Working Document of the Commission Staff – entitled "Implementing Euratom Treaty Safeguards" (IETS) [1] – had by then evolved into a consensus-generating document.

At the November 2006 meeting in Luxembourg, the WPAQ experts and the Commission advisory group agreed that the proposed Commission Working Document sufficiently reflected the shared understandings. The Finnish Presidency later congratulated the work of all experts involved, from both parties. The first strategic paper on implementing the Euratom safeguards system, based on shared understandings between the Commission and the Council, was born.

Both the Commission and the WPAQ agreed that these shared understandings would evolve, e.g. subject to the outcome of the discussions with the IAEA, and to the outcome of the audit-like inspection trials conducted on a voluntary basis at some nuclear installations. The document should be seen as a living document. However, both parties also agreed on the benefits of recording the large consensus reached. The general impression amongst the experts was that the Community had, for the first time since long a solid shared position about safeguards in Europe.

Once finalised by the Commission Services, the Working Document was approved internally on the 27 February 2007. The next day, the Council's COREPER took note of the document, sealing this landmark shared strategy text.

The way forward is now open. There will be a need to further discuss and agree on subjects covered by the Working Document, including detailed co-ordination arrangements and site and facility-specific issues within the agreed framework involving all relevant parties, i.e. the Commission, Member States, Operators and IAEA as appropriate.

Since early 2006, the European Safeguards Research and Development Association (ESARDA) created a working group with focus on best practices for nuclear material accounting and control systems and the use of audit methodologies for assessing the assurance provided by such systems, called Nuclear Material Accountancy and Audit Focus Group (NMACAF). As a member of ESARDA, the Commission services attend the meetings of the group. Future updates of the Commission's strategy document intend to take into account the group's outcomes as they become available.

## What's in IETS?

The Working Document of the Commission Staff SEC(2007)293 – intitled "Implementing Euratom Treaty Safeguards" (IETS) – outlines the European Commission's approach for implementing the Euratom safeguards system set forth in the Treaty, reflecting the shared understandings with the Council resulting from a dialogue started in 2005.

The Working Document effectively implements the Commission's strategy for improved nuclear safeguards effectiveness and efficiency in the Community – the revised approaches –. This strategy was described in the Commission Communications of 30 April 2004 [3] and 15 June 2006 [4], respectively on the implementation of nuclear inspection and safeguards tasks devolved to the Directorate-General for Energy and Transport of the European Commission and on the implementation of nuclear safeguards within the European Union.

The Working Document fully incorporates the WPAQ's proposals on a new framework for Euratom safeguards. These proposals were described in the WPAQ's 'non-paper' on "A New Framework for Euratom Safeguards" of 15 December 2005, reproduced in an annex of the Working Document. The mail text frequently refers to specific paragraphs of the WPAQ 'non-paper'.

The Working Document combines the Commission's strategy for improved nuclear safeguards effectiveness and efficiency in the Community and the WPAQ's proposals on a new framework for Euratom safeguards. It records the large consensus reached between the Commission and the Council on their shared understandings about nuclear safeguards in Europe. These shared understandings are underpinned by a small number of simple concepts.

The Euratom safeguards system permits the Community to make certain that nuclear materials are not diverted to purposes other than those for which they are intended. It provides a supervision mechanism for the European Commission to satisfy itself that:

– in the territories of Member States nuclear materials are not diverted from their intended uses as declared by the users;
– the provisions relating to supply and any particular safeguarding obligations assumed by the Community under an agreement concluded with a third State or an international organisation are complied with

The Euratom safeguards system permits also the Community to meet its international safeguards obligations with regards to the International Atomic Energy Agency (IAEA) and certain third States:

– it requires the Commission, through co-operation with the IAEA, to satisfy itself though that source and special fissionable material in all peaceful nuclear activities within the territories of the States are not used for nuclear weapons or other nuclear explosive devices undeclared;
– it provides a system of accounting for and control of nuclear material meeting the requirements agreed between the Community, the Member States and the IAEA

When the Commission administers the Euratom safeguards system for meeting the objectives of Euratom Treaty Article 77a – e.g. as a supervision mechanism – it obtains a triangular set of assurances:

– assurances on the compliance of the nuclear operator with the applicable provisions and restrictions;
– assurances on the performance of the nuclear operator's systems addressing the accounting for and control of nuclear material (NMAC);
– verifications of non-diversion of nuclear material, confirming the credibility of the nuclear operator's declarations;

When the Commission administers the Euratom safeguards system for meeting its international safeguards obligations, it relies on a simple set of key principles:
– The Commission is the single interlocutor of the IAEA for all matters where the Community is competent (e.g. co-operative planning, joint inspections, avoiding duplication of activities, sharing of data and equipment);
– Both the Commission and the IAEA must be able to fulfil their own safeguards obligations and to preserve their capacity to reach their own independent conclusions
– The Commission and the IAEA co-operate in compliance with the verification agreements signed in 1973, 1976 and 1978, including the Protocols additional to those agreements, signed in 1998;

In either case, for implementing the Euratom safeguards system, the European Commission uses controls performed either at headquarters or during inspections at the nuclear installations. These controls should be an effective and efficient combination of:

– independent physical verifications, e.g. using analytical tests like nuclear measurements;
– NMAC system assessments, e.g. using audit methodologies

A significant part of the Working Document details and discusses enhancements to the modalities for cooperation with the International Atomic Energy Agency (IAEA) on nuclear safeguards matters. A review of these enhancements is outside the scope of this paper.

The Working Document is further supplemented by technical annexes that provide:

– Detailed information about the key control activities that the Commission considers for meeting the objectives of Euratom Treaty Article 77a
  o Performance control activities (both on-site and at headquarters), including audit-type inspections
  o Verification activities (both on-site and at headquarters)
  o Inspection activities cutting across the three types of control activities above, depending on the type of installation under consideration (out of six types: reactors, fuel fabrication, enrichment, reprocessing, storage and small installations)
– The preliminary NMAC systems and process model underpinning the Commission services' activities for making certain, by appropriate supervision, that nuclear material are not diverted to purposes other than those for which they are intended.

## Conclusion

The Euratom Treaty outlines a nuclear safeguards system with multiple objectives aimed at ensuring a safe and secure use of nuclear material in Europe. The Community's agreements with the IAEA and third parties

provide for a system of safeguards which partially overlaps the objectives of the Euratom Treaty.

The Commission Staff Working Document SEC(2007)293 – entitled "Implementing Euratom Treaty Safeguards" (IETS) – of 27 February 2007 was taken note of by the Council's COREPER on 28 February 2007.

The Working Document identifies clearly the different but linked objectives of the Euratom safeguards system, whether serving as a supervision mechanism for the Commission. Although not new, this clarification contributed to a faster convergence of views during the discussions between the Council and the Commission, as it allows to better separate the concerns.

The text of the Working Document, and the shared understandings that it reflects constitute the main achievement of the dialogue set up in 2005 between the Council WPAQ and the Commission services.

One achievement of the dialogue between the Commission and the Council is having generated a joint vision of the roles of the various actors involved. This vision confirms the role of the Commission as the single interlocutor for external bodies, but also asks for further debate about how the national inspectorates should contribute to the overall system.

The process of dialogue within the Community between the Commission and the Council has now moved into a new period. Having established a regular dialogue between the Commission services and the Council WPAQ allowed the Community to reap immediate benefits. The Commission's Working Document is but one of them.

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# French experience in the implementation of the Additional Protocol

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***Abstract:***

*When the French Additional Protocol entered into force on April 30th 2004, the French authorities (CTE) and their technical support (IRSN) had already carried out some actions in order to meet the new obligations of this agreement. As this protocol extends the scope of the Agency's control to operators (Nuclear fuel cycle related R&D, manufacturing and export of some specific equipment and non nuclear material, ten years nuclear fuel cycle related R&D plans …) which do not manipulate nuclear materials, a tremendous work was done to be able to send in due time the initial declaration to the I.A.E.A..*
*Three main activities were carried out in parallel:*

1. *Identification of the entities potentially concerned,*
2. *Elaboration of a declaration handbook,*
3. *Elaboration of declaration forms.*

*Once the entities potentially concerned by a declaration under the Additional Protocol were identified (more than 1700), the French authorities informed them of the context and of the obligations of the Additional Protocol and sent them a declaration handbook so that they could check whether they were actually concerned or not.*
*All the entities answers were analysed by IRSN. The main part of the work consisted in giving support to the companies on the redaction of their declarations, in verifying that all the information required by the declaration forms were provided and in crosschecking and consolidating the declarations made by different actors of a common R&D program.*
*This work enabled the French authorities to submit in time the initial declaration under the Additional Protocol.*
*The French annual declarations for 2004 and 2005 were updated by following a very similar process to the one described above with some improvements.*

**Keywords:** French Additional Protocol, initial declaration, entities, safeguards.

## 1. Introduction

In 1992, France ratified the Nuclear Non-Proliferation Treaty, which it regards as the cornerstone of the international non-proliferation regime.
France supports the central role of the safeguards system of the International Atomic Energy Agency (IAEA). Within the safeguards system, the implementation of an additional protocol, together with the implementation of a comprehensive safeguards agreement is essential in order to provide complete certainty that non-proliferation commitments under the Treaty are being respected.

On September 22nd 1998, in order to participate in the strengthening of IAEA safeguards, France signed an additional protocol to its safeguards agreement. The protocol entered into force at the same time as those of the other members of the European Union, on April 30th 2004. Under that protocol, France will provide additional information to IAEA, for example on activities carried out in cooperation

with non-nuclear-weapon States (NNWS) in connection, inter alia, with research and development work on the nuclear fuel cycle (whether nuclear materials are involved or not). Exports of certain types of equipment and non nuclear materials (annex II) will also be declared regularly to IAEA.

The purpose of this commitment is to facilitate the detection of undeclared nuclear activities in a non-nuclear-weapon State. To this end, IAEA may also request complementary accesses to French installations.

Concerning the organisation put in place on a national level for the implementation of the additional protocol in France, the "Comité Technique Euratom" (Euratom Technical Committee, CTE), which depends from the prime minister services and is, inter alia, responsible for relations with IAEA and the European Commission with regard to the implementation of international controls on nuclear materials within French territory, is the French authorities representative.

In the accomplishment of this task, the CTE relies on the technical support of the Department for the Implementation of International Safeguards (SACI) of the Institute for Radiological Protection and Nuclear Safety (IRSN). This institute carries out research and provides expert advice in, inter alia, the areas of the protection and control of nuclear materials.

France did not delegate AP duties to the European Commission (non-side letter country). Only declarations linked with two articles (2.a.iv) and 2.a.v)) concerning nuclear materials (mines, conversion plants, source materials like ore concentrates) are made by the European Commission who transmits it to IAEA and send a copy to the French authorities.

## 2. Preliminary stage

Between the fourth quarter of 2003 and the second quarter of 2004, CTE and IRSN began preparing the implementation of the Additional Protocol. During this preliminary stage, three main activities were carried out in parallel: elaboration of a specific declaration handbook and declaration forms, identification of the entities potentially concerned and organisation of meetings with the major public operators.

### 2.1. Elaboration of a specific declaration handbook and forms

The specificities of the French Additional Protocol - no site declaration, declaration of activities carried out in cooperation with NNWS …- made the guidelines produced by the IAEA not directly usable by the French entities potentially concerned by this agreement. For this reason, it was decided to create specific national declaration forms and the declaration handbook associated specially designed to fit with the French Additional Protocol requirements.
Seven specific national declaration forms were created corresponding to the articles of the French Additional Protocol:

1. Identification of the entity,
2. State controlled nuclear fuel cycle-related research and development (Art. 2.a.i)),
3. Private nuclear fuel cycle-related research and development (Art. 2.b),
4. General plans for the succeeding ten-year period relevant to the development of nuclear fuel cycle (Art. 2.a.viii)),
5. Import or export of nuclear material exempt of safeguards (Art. 2.a.vi)),
6. Description of scale of operation engaged in the activities specified in Annex I of the Protocol (Art. 2.a.iii)),
7. Export of equipment or non nuclear material listed in Annexes I or II of the Protocol (Art. 2.a.vii)a).

These forms were created under Ms Excel© and provided to the entities potentially concerned under electronic and paper format.
Each entity was required to fill the identification form and the forms corresponding to the activities to be declared.

As a complement to these forms, a national declaration handbook was also elaborated. This document is made of three parts:

1. The first part named “are-you concerned ?” was elaborated in order to help the entities potentially concerned to identify quickly, through synoptic tables, if there activities were subject to declaration or not,
2. The second part presents the different types of declarations (initial, annual updates or quarterly) and the timetables of submission,
3. The third part explains how to fill the different declaration forms by providing definitions and several examples.

### 2.2. Identification of the entities potentially concerned

As the Additional Protocol covers activities which may not involve nuclear materials, it was not an easy task to identify the French entities potentially concerned as they were not limited to those controlled by the European Commission or the IAEA. At that time (end of 2003), no specific data base of these companies was existing. For this reason, CTE and IRSN decided to use different information sources to create the list of entities potentially concerned. This list was constituted of several parts depending on the source of information used:

1. The first part of the list was based on a request made on the Kompass© data base[1] following several criteria. The criteria chosen were activities directly or indirectly in connection with nuclear research and development and activities or equipment in connection with those listed in Annexes I or II of the Additional Protocol. The list obtained was analysed to remove the companies which seemed not relevant. After this process, the first part of the list contained 1484 entities,
2. The second part of the list was elaborated thanks to the information provided by the French export control division of the French Ministry of Industry,
3. The third part of list contained the companies owning nuclear materials which were already subject to Safeguards.

At the final stage, the list included 1741 entities potentially concerned.

### 2.3. Meetings with the major operators

In parallel to the elaboration of the declaration forms and handbook and the identification of the entities potentially concerned, CTE and IRSN organised several meetings with the main actors of the French nuclear sector (EDF, AREVA) and nuclear public research organisation (CEA, CNRS).
The aims of these meetings were to present the French organisation for the implementation of the Additional Protocol, to explain the various types of declaration and to identify coordinators inside these entities.

## 3. Collecting and processing the declaration information

The preliminary stage finished at the end of April 2004 with the sending of the declaration files containing a letter explaining the aims of the Additional Protocol, the declaration forms and the declaration handbook, to the 1741 entities identified.
An answer was requested for July 15th 2004.

### 3.1. Collecting the information

The answers were received by IRSN in several phases between April and September 2004.
At the beginning the rate of answer was increasing very slowly and the major part of the answers were negatives. When reaching the deadline, on July 15th 2004, only 44% of the entities contacted had provided an answer. As a consequence, CTE and IRSN decided to send a letter of recall to entities which had not yet answered. This procedure was very efficacious (+19% of answers in 15 days).
At the end of the collecting process in September 2004, 1382 entities (79.4%) had provided an answer to IRSN.

[1] Kompass© is a commercial data base which references the companies based on several criteria.

Between May and September 2004, IRSN implemented a hot line especially dedicated to assist the operators who needed help to analyse their situation regarding the declaration criteria or needed help to fill the declaration forms. This specific phone number was used by many operators.

### 3.2. Processing the information

IRSN analysed the information provided in the 1382 answers.
For the 1354 negative answers, the work consisted, when it was possible, in analysing the information provided by the entity to justify that it was not concerned, in order to verify that there was no misunderstanding.
For the 28 positive answers, the analysis was more complex. It consisted in:

1. Verifying that the information provided was exhaustive,
2. Verifying that the activity declared was done consistently with the scope and objectives of the French Protocol and in accordance with the relevant articles of the Additional Protocol,
3. Consolidating the declarations of the same R&D program made by several French entities involved.

In many cases, the analysis of the answers leads IRSN to identify new entities and to contact them to ask them if they were subject to declaration or not.

As a result of the process of the information, IRSN provided CTE with a draft version of the French initial declaration.

## 4. The French declarations for the Additional Protocol

Based on the draft version of the French initial declaration prepared by IRSN, the CTE made the final decisions in order to validate the final draft version.

The next step consisted, for CTE, in sending this draft declaration to the ministries concerned by the AP implementation to get their validation or remarks.

Finally, after having taken into account the remarks received from the ministries, CTE finalised the official French declaration which was sent to IAEA by France's representative on the Board of Governors of IAEA.

A similar process is followed for annual updates and quarterly declarations.

The workload of the implementation of the AP at the national level, especially for the preparation of the initial declaration, is worth to be noted: 1 man.year for IRSN to which must be added the work carried out in CTE.

### 4.1. Initial declaration

The initial declaration was sent in due time to the Agency on October 27$^{th}$ 2004. It represented a document of about 300 pages with the following main contents:

| | |
|---|---|
| State-controlled R&D (Art. 2.a.i)) | 262 entries (13 entities declared) |
| Scale of operations for annex I activities (Art. 2.a.iii)) | 9 entries (8 entities declared) |
| Ten year plan (Art. 2.a.viii)) | 34 entries (8 entities declared) |

Altogether, 22 different entities made a declaration in the frame of the initial declaration.

All the R&D cooperation with NNWS were declared as "state controlled" considering that the programs were in every case at least known and authorised by the French authorities.

### 4.2. Annual declarations

For the preparation of annual declarations, the French authorities write each year to a selected list of entities (199 for 2004, 189 for 2005). Based on the feedback experience of the initial declaration, an

effort was made in order to ask the entities to improve the precision of their declaration (description of activities, location of activities, reasons for managed access,…).
For the 2005 declaration, the list was built as follows:

- Entities which actually made a declaration the previous year,
- Entities which did not actually made a declaration the previous year but which carry out activities potentially concerned by the AP,
- Entities which did not officially declare the previous year that they were not concerned by the AP,
- Entities which did not answer to the French authorities the previous year,
- Entities newly identified as potentially concerned by the AP.

About 75 % of the entities consulted answered by informing that they were not concerned or by sending a declaration.

The 2004 and 2005 annual declarations had the following main contents:

| | **Number of entries** | | **Number of declaring entities** | |
|---|---|---|---|---|
| | **2004** | **2005** | **2004** | **2005** |
| State-controlled R&D (Art. 2.a.i)) | 302 | 320 | 11 | 12 |
| Scale of operations for annex I activities (Art. 2.a.iii)) | 9 | 8 | 7 | 6 |
| Ten year plan (Art. 2.a.viii)) | 60 | 47 | 9 | 10 |
| Total for the all declarations | 374 | 381 | 20 | 19 |

### 4.3. Quarterly declarations

The quarterly declarations are required for the exports of equipment and non nuclear materials to a Non Nuclear Weapon State (NNWS). For their preparation, the French authorities write each year and each quarter to a list of entities (about 55 for 2005 and 2006). This list is established as follows:

- Entities who actually declared an export in the previous year or during the last quarter,
- Entities who did not declare an export in the last declaration but which carry out activities potentially concerned,
- The information provided by the French export control division of the French Ministry of Industry.

To respect the appendix III of the additional protocol, the produced declarations are split into two parts, taking into account whether the export is made towards a NNWS member of the European Union or not.
France thus produces two declarations by quarter. Additional declarations can also be produced during the year to complete the initial statements (declarations) of every quarter (10 declarations were produced in 2006).

## 5. Conclusion

The Additional Protocol implementation in France represented a heavy workload for the French authorities (more than 1 man.year), as well as for the declaring entities, especially for the initial declaration. Nevertheless, this declaration was sent in due time and both annual updates and quarterly declarations are carried out smoothly. The “quality” of the declarations is improved each year and is now reaching an asymptote!

The result of this effort was that a significant amount of information was made available to the Agency, in particular on the R&D cooperation programs of French entities with NNWS
Until now, the declarations sent to the IAEA were established in the "text" format. The French authorities are studying the possibility of using the “Protocol reporter” software developed by IAEA for the next declarations. A test is foreseen for the 2006 annual declaration.
The aim is to facilitate the Agency’s work when processing the declarations.

# Facing the new challenge of the Additional Protocol among the firsts
# Hungarian experience in the Additional Protocol implementation[1] [1]

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***Abstract:***

*Hungary signed the Additional Protocol (AP) in 1998. After ratification by the Parliament, the protocol entered into force in April 2000. Implementation of the Additional Protocol represented a new challenge for both the International Atomic Energy Agency (IAEA) and the Hungarian stakeholders, since none of them had experience in this field. First declaration under AP was sent to the IAEA in October 2000. Since then, seven up-dated annual declarations have been sent to the IAEA. Verification of our declarations by the IAEA was intensive in the first two years, later number of complementary accesses (CA) has been decreasing.*

*The first conclusion on the absence of undeclared nuclear materials and activities in Hungary was drawn and published in the IAEA's SIR in 2003. Introduction of integrated safeguards was preceded by a successful unannounced inspection trial at the Paks NPP. As a result of an intensive co-operation between the Hungarian SSAC[2], the Hungarian installations and the IAEA, implementation of integrated safeguards system (IS) started in November 2004. IAEA's integrated safeguards approach models for different types of nuclear installations were tailored for the Hungarian installations.*

*Hungary joined the European Union (EU) and Euratom Treaty entered into force in May 2004. By that time implementation of the Additional Protocol was already underway in Hungary. INFCIRC/193 was ratified in Hungary at the end of 2006. By that time integrated safeguards was already applied in the country.*

*This article intends to cover some of the major issues encountered:*
- *experience in the implementation of the Additional Protocol*
- *applying integrated safeguards measures in the Hungarian installations*
- *switching to the Euratom System (Hungary as a non-side letter state)*
- *maintaining the Hungarian SSAC in addition and in complementation to the Euratom System.*

**Keywords:** additional protocol; integrated safeguards; Hungary

## 1. Introduction

From the very beginning Hungary has been committed to the international non-proliferation regime and the disarmament initiatives. Hungary joined the Treaty on the Non-Proliferation of Nuclear

[1] General disclaimer**:** the views expressed in this paper are the individual views of the authors, and are not meant to represent the official view of their organisations
[2] State System of Accountancy for and Control of Nuclear Materials

Weapons (NPT) in 1968 (entered into force in March 1970) [2] and signed a comprehensive Safeguards Agreement[3] (entered into force in March 1972) on the verification regime of the NPT obligations. [3] In the early 90s traditional methods of safeguards system faced new challenges. After the 1st Gulf War the clandestine nuclear programme of Iraq revealed, and in the middle of the decade the North-Korean (DPRK) undeclared nuclear activities were brought to light. [4] These events forced major revision of efficiency and credibility of the IAEA safeguards system. The principal legal document of the strengthened safeguards system is the Additional Protocol (INFCIRC/540), which was signed by Hungary in 1998 and entered into force in April 2000, among the firsts. [5]

The traditional safeguards is based on regular ad hoc and routine inspections of declared nuclear materials and installations of the State Signatories. The AP new verification strategy emphasises the unannounced and short-notice inspections of nuclear activities and capabilities, and is not limited to declared nuclear installations and materials only. The IAEA requires declarations concerning all of the nuclear related activities (e.g. research), descriptions of buildings on sites, historical data, etc. The provisions of AP authorized the IAEA to perform complementary accesses and use broader verification methods (environmental sampling, satellite imagery etc.). The IAEA also recognised the high added value of the collection and evaluation of the open-source information in support of its verifications. These new elements provide really efficient tools to IAEA for early detection of any kind of irregularity and violation of the NPT, especially to reveal the undeclared nuclear materials and activities.

After the enlargement of the European Union the regulatory environment became more complex. The Hungarian safeguards system must fulfil the obligations required by the Treaty establishing the European Atomic Energy Community (Euratom) from 1st of May 2004. [6] From this date provisions of the AP have been implemented in EU as well. The trilateral Safeguards Agreement (INCIRC/193) is expected to enter into force in Hungary on 1 July, 2007. [7] The new safeguards regulation in Hungary [8] is fully compatible with all principles and provisions of the Commission Regulation (Euratom) No. 302/2005 on the application of Euratom safeguards. [9]

In November 2004 a special verification regime of IAEA has been introduced in Hungary. Hungary was the first country with significant nuclear activities where the new integrated safeguards system was introduced.

## 2. Nuclear Fuel Cycle in Hungary [10]

The nuclear fuel cycle in Hungary is not closed, both ‘end’ of the cycle is missing. There is no more active uranium mine (it was closed in 1997). Currently, in the location of the closed down mine (located in the south-west of the country, in the Mecsek mountains), a mine-water treatment installation is in operation. Their activities are restricted to site maintenance and remediation. This installation produces 2-3 tonnes/year of uranium ore extraction (‘yellow cake’), as a side-product of the water cleaning processes (less than 1 effective kg annually). The whole production is exported to France in every 2-3 years. Hungary has never acquired conversion or fuel fabrication capability and no reprocessing activities are foreseen.

[3] INFCIRC/174 based upon the model INFCIRC/153 type

**Figure 1:** closed-down uranium mine in Hungary

The reactors are in the core of the Hungarian nuclear activities. There are research and training reactors in Budapest and power reactors in Paks (about 100km to south of Budapest on the right bank of Danube river). The Budapest Research Reactor (operated by Hungarian Academy of Sciences KFKI Atomic Energy Research Institute) was established in 1959. After certain reconstructions the thermal power of the pool type reactor is 10 MWth. The fuel of the reactor is 36% high-enriched uranium (HEU). There is fresh fuel storage and in- and out-door spent fuel storage as well. The principal activities are the research and isotope production. The 100 kWth pool type training reactor (operated by the Budapest University of Technology and Economics, BME) was put in operation in 1971. The fuel is 10% low-enriched uranium (LEU).

**Figure 2:** Budapest Research Reactor

**Figure 3:** Training reactor BME

The power plant (Paks NPP) includes 4 units of WWER-440 PWR reactors, which are located in 2 separate halls. The units started their operation between 1982 and 1987. These Russian type reactors' nominal electricity power is 440 MWe, but nowadays the electric output is increasing up to 500 MWe (currently, about 460-480 MWe). The power plant has 4 spent fuel ponds (one for each reactor unit) and 2 fresh fuel storage (for each reactor hall). For interim storage of the spent fuels of the power plant, an on-site Modular Vault Dry Storage was constructed in 1997. Currently, it is separated by fence from the power plant and operated by a different company. The modular structure provides very flexible capacity and the facility is planned to store for 50 years all the spent fuel assemblies coming from the power reactors during their life-time. Concerning the quantity of the nuclear material, this installation is the most significant in Hungary.

**Figure 4:** Paks NPP

**Figure 5:** Interim storage facility for spent fuel assemblies

The rest of Hungarian activities is very limited concerning the amount of the nuclear materials or sensitivity of the technology, but very various geographically. There is a small isotope storage belonging to the Central Research Institute for Physics in Budapest and in many other locations country wide there are small users and owners of the nuclear materials (universities, research institutes, industrial companies etc.). In many cases the nuclear materials are not used anymore, therefore the idea of collecting them arose recently to increase of the control (safety and security) and decrease the risk, vulnerability and problems.

## 3. Additional Protocol

The eight years, which elapsed since the ratification of the Additional Protocol, seem to be really long and it was rich with new challenges and experiences. Hungary was involved very early in the development of Strengthened Safeguards through the Hungarian Support Programme's participation in tasks (e.g. environmental monitoring) related to the IAEA's Programme 93+2, on which the new system is based. Australia was the first state to bring the AP into force. In the application of the new measures of the Additional Protocol, Hungary was a kind of pilot country.

The Article 2 of AP defines the scope of provision of information. In October 2000 the initial declaration was sent to IAEA by Protocol Reporter software. The Protocol Reporter was developed by the IAEA to assist States in establishing a database, in processing, maintaining and updating information, and in providing declarations to the IAEA in electronic format.

Collection of all necessary information under Article 2.a.(i) regarding nuclear fuel cycle-related research and development activities from the potential data providers, much beyond the nuclear material licensees and facility operators demanded an outreach programme involving Hungarian Academy of Sciences and Ministry of Education as well. In accordance with the provisions of AP the future plan of R&D activities must be submitted to IAEA as well.

AP obligations demanded a site declaration involving all buildings on site as well. Preparation of this declaration required really accorded efforts of the Hungarian Atomic Energy Authority (HAEA) and the operators as well. Defining the site boundary is the State authority's obligation. It was understood that the boundary should realistically represent the limits of the work site, with no areas artificially or arbitrarily separated by fencing to limit verification access. The Agency's right to perform complementary access depends on whether a building is or is not on a 'site'. For all buildings on a site, the Agency has strong access rights to 'assure the absence of undeclared nuclear material or activities'. 'Site' means an area surrounding a facility or location-outside-facility (LOF). Several facilities or LOFs may share a site. Sites should normally not be smaller than one building, and must include 'installations co-located with the facility or LOF for the provision of essential services', such as: hot cells, waste treatment, Annex I activities. It was also understood that if the Agency considers that a building not within a site might be 'functionally related' to the site, the Agency can request declaration of the building under Article 2.b.(ii). Examples could be buildings with related scientific activities, used for fabricating equipment, housing for computer services or personnel. Difficulties arise if a building is not included in a site, and there is no initial declaration so the Agency does not know what the building is and cannot judge whether it is 'functionally related'. With this understanding Hungary did its best to define its sites under the AP.

The Hungarian definition of the sites of the AP did not follow the former structure of the 7 Material Balance Areas (MBA)[4].

| MBA code | EURATOM code | Site name and location |
|---|---|---|
| HU-A, HU-D, some KMPs of HU-C | *WHUA, WHUD, some KMPs of WHUC* | Campus of Central Research Institute for Physics (KFKI) |
| HU-B | *WHUB* | Training Reactor, Budapest University of Technology and Economics |
| HU-C | *WHUC* | 7 sites, each belonging to HU-C |
| HU-E, F, and G | *WHUE, WHUF, WHUG* | Paks (NPP and MVDS) |

**Figure 6:** The Hungarian AP sites

In Hungary the most critical site definition was the campus of Central Research Institute for Physics where several facilities and installations (3 MBAs, several KMPs and isotope production, nuclear research, hot cells, laboratories etc.) share the sites with a lot of non-nuclear and non-research companies and buildings. At that time Hungary did not insist to define 'island' in this site. Preparation of the correct site declaration required additional efforts from operator's side because in case of several buildings of non-nuclear use the owners and functions of the buildings change very often. A site representative was appointed for this site to collect the relevant data.

Paks site under AP comprises the four units of the Nuclear Power Plant, as well as the Modular Vault Dry storage. Collection of data for site declaration is well organized and represents a good example of department to department co-operation within the NPP.

Under Article 2.a.(vii) of AP the exempted material became the subject of declaration and control. Unfortunately, in the former practice, exemption (EQ, EU) was requested very often for the used up material instead of the appropriate code (TU). Tracking of this historical data and materials was quiet difficult: some of the items and information could not be found. Finally, full physical inventory taking

[4] The newly created 8th MBA (WHUH) – closed uranium mine – is not defined as a site

was necessary. The owners of exempted materials were basically unaware of safeguards obligations because of the lack of former experiences, therefore their training for this new task is an important issue.

Obligations laid down in the AP cover the uranium mines, which were out of scope of traditional safeguards system. By the time the AP entered into force in Hungary its uranium mine was closed down. The present activities on the site are restricted to site maintenance and remediation. As a by-product of the water treatment activities, uranium ore concentrates are extracted in small quantities – less than 1 effective kg annually – which is usually exported.

The holdings and movements of its side-product ('yellow cake') as a source material under Article 2.a.(vi) must be reported, but it is not subject to verification. However, some complementary access inspections (advance notice at least 24 hours before) were carried out by IAEA in the closed uranium mine of Hungary to verify the declared nuclear activities and the absence of any undeclared activities. The usual tools of these CA inspections are the environmental sample taking[5], detailed design information verification by visual observation[6], radiation monitoring, screening historic data and information and a general overview of the current activities and future plan.

Hungary made certain efforts to collect all of the relevant historic information and data concerning the past nuclear activities: in 2006 a study was prepared on the former application of nuclear materials. It was necessary since the pre-safeguarded time period was not well documented and the fundamental source of information relied on the memory of elder personnel, and even that personnel disappeared mostly.

The CA inspections are important verification methods in the strengthened safeguards system. They are performed either with a 2 hour advance notice in conjunction with traditional safeguards inspections or with 24 hour advance notice. General rule for CA inspection: it must be performed in week-days and working hours. The smooth access requires flexibility and special internal measures on the operator side. In the first two years the number of CA inspections was relatively high (10 times annual), then from 2003 this number reduced significantly: in the last years about 3 times annual is constant. After the first years (access with 24 hour notification in LOFs, exempted materials etc.) 2 hour notice accesses are mostly performed.

The Hungarian policy regarding AP issues is very flexible and co-operative. There was no managed access requested in course of the complementary accesses. Article 8 access was not requested by Hungary. Up to now neither Hungary nor the IAEA indicated that Subsidiary Arrangement to the AP was needed.

The 7$^{th}$ sets of AP declarations have already been submitted to the IAEA. With using Protocol Reporter software the Hungarian declarations are complete, well structured, quickly submitted and ready for easy evaluation.

In conclusion we can state that the implementation of the AP in Hungary in the last 7 years was smooth and a good example of a fruitful co-operation between an SSAC and the IAEA.

## 4. Integrated Safeguards

The fruit of the hard work as pioneers in the implementation of the AP was the favourable final conclusion on the absence of undeclared nuclear materials and activities in Hungary drawn by the IAEA and published in the SIR in May 2003. With this conclusion Hungary and the IAEA could started to discuss the conditions of an Integrated Safeguards (IS) System in Hungary. The IS is an advanced method of the most comprehensive and efficient verification regime. The basic concept of IS is the expanded use of unannounced inspections within the scheduled routine inspection regime. [11] The strengthened safeguards system shifts emphasis from declared nuclear materials to all related activities and capabilities and from mechanistic verification to acquiring wide range of information and access, and using transparency as well. Enabling unannounced inspections requires special efforts

[5] The high performance trace analysis (HPTA) evaluation method based upon the mass-spectrometry
[6] Taking pictures generally allowed in Hungarian installations, but the operator has the right to deny it

and capabilities of HAEA and installations as well, but there is no doubt concerning its high relevance in safeguards, especially in the early detection of undeclared nuclear materials and activities.

The IAEA prepared facility specific IS for each installation and tailored the model approach to the Hungarian facilities. The first trial unannounced inspection was held in Paks NPP in October, 2003. After the careful evaluation of previous experience, in November 2004 the implementation of IS was started. IAEA made special arrangements on IS activities in Hungary. These arrangements cover the verification activities and safeguards measures (CA, environmental sampling etc.) and the requested operational information. The arrangements also define the inspection notification procedures. The description of IS activities are subject of revision. In WHUB, WHUC and WHUD MBAs the installation specific IS activities are very limited: only one physical inventory and design information verification[7] (PIV/DIV) in each facility every 4-6 year. There are no surveillance and containment. These 'weak' measures are justified by very small quantity of nuclear material and almost static inventory (especially in WHUB and WHUD). In WHUA one PIV/DIV and at least one announced inspection in every calendar year means the appropriate IS measures (because of higher grade and bigger quantity of nuclear materials and characteristic of activities). There are no surveillance installed, but the metal seals are applicable for fresh fuels and for spent fuels, if necessary.

Certainly, the IS measures are stronger in that installations where the amounts of the nuclear materials and the typical activities are significant. In Paks NPP (WHUE and WHUF) one PIV/DIV and at least one unannounced inspections are required in every calendar year. Instead of core control verification (typically uncertain timing) one pre- and one post-core-loading verification inspections (in the regular working hours) are defined. These inspections include verification of fresh and spent fuels as well. With regard to the containment and surveillance applied: VACOSS seals (can be either attached or detached by the operators) and metal seals are applicable, surveillance is running in overwrite mode. In case of open core back-up surveillance is installed.

Finally, in the interim storage of spent fuels (WHUG) the measures follow the working loads of the installation: during the spent fuel transfer campaigns inspections are carried out for sealing the new receipts (with COBRA and metal seals). Fuel monitoring and surveillance systems are applied as well. The annual PIV/DIV and at least one unannounced inspection are regular.

In the IS system the operational information is highlighted, therefore the required advance information is strictly defined. Basically, IAEA must be informed about all relevant information: shipments, receipts, reactor shut-down, spent fuel pond opening/closing, spent fuel cask movements and fuel handling machine transfers. The planning of unannounced inspections for IAEA requires nearly daily communication. In this communication the HAEA is a crucial interface. All of the relevant information must be forwarded to IAEA which has any kind of effect to the performance of an unannounced inspection: e.g. public holidays.

Proceeding unannounced inspections (UI) was a really sensitive part of the implementation of IS in Hungary. Finally, the arrangements covered all of the essential aspects of the unannounced inspections. The access will be granted within 2 hours of the announcement of the inspection. The designated IAEA inspector can start the inspection (visual observation) without the presence of national inspector. The inspectors carry a letter authorising an UI. The access will be granted at an agreed gate of the installation and the entering process can be faster and simpler if the inspector's personal data are provided previously to the facility.

The maximum efficiency of the strengthened safeguards system needs the universality of AP. Hungary is open to share its own experiences with other countries and help them to meet the requirements. Regional meetings and ESARDA provide great opportunity for it. In the framework of the Hungarian Support Programme to the IAEA safeguards Hungary organises trainings and meetings in the AP, CA, IS and UI issues. The Hungarian contribution in the technical support was also significant: participation in developments of new safeguards verification methods (e.g. SFAT for non-fuel items, environmental monitoring).

## 5. Together with the European Commission – Participation in Euratom System

[7] In the Euratom terminology: basic technical characteristic (BTC) verification

At the time of the first enlargement of EU, on 1 May 2004, the provisions of AP have already been implemented in Hungary. AP entered into force in EU on the same day. After joining EU the legal structure of the Hungarian safeguards became more complicated. The Euratom Treaty [6] and the Safeguards Regulation [9][8] entered into force immediately, but the ratification of the trilateral Safeguards Agreement [7] was completed only at the end of last year and the implementation would start in July 2007. Hungary is considered as non-side letter state, which means that the competences and authorisations will be shared in respect of the implementation of AP between the national and supranational regulatory authorities. The European Commission (DG TREN) is responsible for the implementation of safeguards measures if any nuclear materials are involved (mines, source materials, exempted materials and waste). SSAC in Hungary is competent in the R&D issues, technology, development, export/import and manufacturing of equipments. For declarations under 2.a.(iii) the AP stipulates a joint responsibility (Community and Hungary). The HAEA and the EC jointly define each site subject to reporting under the AP. Hungary designates a site representative for each site from its staff. Under the Article 2.a.(v), (vi) and (vii) the operators' declarations will be sent directly to EC and parallel to SSAC. The AP stipulates a joint responsibility (Community and Hungary) for the provision of information on nuclear waste (Article 2.a.(viii)).

The Hungarian national safeguards regulation [8] is fully compatible with the principles and provisions of the Euratom Treaty and the EC Safeguards Regulation. The independent Hungarian Atomic Energy Authority maintains SSAC, and has its own right to perform independent inspections and participate in all of the international/supranational nuclear inspection under the jurisdiction of Hungary. HAEA is responsible for reporting of WHUC (catch all MBA). In the implementation of strengthened safeguards measures the SSAC has a key position. It enjoys the benefits of the local and historical knowledge (geography, language, rules, customers, access) and the closer contact with the operators/owners. It plays a very important and significant role in CA and unannounced inspections. The SSAC is also able to provide completeness of information concerning sites, historical data, nuclear related activities not containing nuclear material and gives clarifications to the IAEA's questions.

From July 2007 the new chapter of Hungarian safeguards will be started. The EC and IAEA will perform common inspection instead of previous parallel inspections. The containment and surveillance (C&S) will be under common supervision. The IAEA seals will be replaced systematically by common seals. The evaluations of the C&S results will be performed in Luxembourg, in the Euratom headquarter. All of the accountancy data of nuclear material will be submitted to IAEA by EC. The new MBA codes will be used universally. Fortunately, all players are very well prepared to manage these changes and the added value of Hungarian SSAC will be really high.

The efficient and valuable safeguards system requires good co-operation of all parties. HAEA can provide essential support to EC and IAEA in operating their systems and reaching the common aim: the principles declared in the NPT.

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# Nuclear safeguards and security call for cooperation

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***Abstract:***

*Nuclear safeguards and nuclear security have a lot in common: their objectives are mutually supportive and many of safeguards methods—accountancy, reporting, containment, surveillance, technical verification—contribute directly to security. We suggest that a system integrating the two together will be more than a sum of its parts. Moreover, a strong state system combined with global international cooperation will be effective in fulfilment of the obligations set by international agreements with intent of improving nuclear security, non-proliferation control and counter-terrorism coordination.*

**Keywords:** safeguards, security, state level approach, cooperation

## 1. Introduction

Nuclear safeguards are undergoing a change. The Non-Proliferation Treaty was agreed on 50 years ago, as was the Euratom Treaty. In 2007 the international organisations, IAEA and EC/Euratom, will celebrate their 50 year birthdays.

In half a century technical and technological changes have been enormous and the effects for the industry, also for nuclear industry, have been remarkable. The threat of proliferation of nuclear weapons still exists but the most probable risk scenarios are something else, such as those related to radiological dispersion devices. The interests of nuclear security are integrating strongly with nuclear safeguards.

## 2. International safeguards

The events in Iraq in the beginning of the 1990's gave a boost to the IAEA strengthened safeguards. Environmental sampling and surveillance methods such as satellite imagery had a crucial role in the success of the IAEA investigations. The European Union, in the meanwhile, has grown into 27 states, and the role of Euratom Safeguards has developed accordingly. Euratom is the focal point of the traditional safeguards for the member states and the IAEA, and the two organisations are in the process of fine-tuning their partnership approaches. The member states of the European Union and the parties of the NPT deserve to experience the benefits of fruitful cooperation between the two international safeguards organisations.

The scale of nuclear industry and other nuclear activities varies by state, as do the associated security and proliferation risk scenarios. The safeguards and security approach therefore, must be tailored to meet the needs of each state. Definition of the roles and responsibilities is a first step and a prerequisite for successful safeguards implementation, recognising at the same time that the inalienable responsibility for safeguards and security rests with the state.

### 2.1. IAEA integrated safeguards

The IAEA integrated safeguards is a combination of the traditional safeguards based on the agreement INFCIRC/153 and the strengthened safeguards based on the Additional Protocol to the agreement. The objective of the IAEA integrated safeguards is to verify that there are no undeclared nuclear activities in the state. The integrated safeguards are also supposed to be an efficient and economic way for the IAEA to implement its safeguards. The state safeguards system should be designed to contribute to this goal, together with the Euratom regional system.

### 2.2. New Euratom safeguards

Euratom safeguards system is being renewed along with the international safeguards. The new system is more flexible than the previous one with regard to allocation of resources and effort: safeguards activities may be focussed, intensified and reduced according to the previous inspection results and forecast risk assessment of facilities. Thus, new Euratom safeguards may profit of the full use of the existing safeguards systems of the operators and the states. Within the traditional safeguards regime, the Euratom system remains the focal point, common for all the EU member states.

## 3. State level safeguards approach: present and future challenges

The state systems—nuclear operators and national authorities—carry the responsibility of non-proliferation control and also the implementation of the Euratom safeguards. The Finnish national system endeavours to efficiently enable the IAEA integrated safeguards, implement the new Euratom safeguards approaches, and fulfil the obligations of the UN Security Council Resolution 1540, against proliferation of biological, chemical and nuclear weapons.

An effective state system shall be able to verify the completeness and correctness of its declarations. While accountancy and reporting as well as auditing the operators' safeguards systems will stay at the heart of safeguards, technical verification and surveillance methods —NDA, environmental sampling, satellite imagery—as well as non-routine inspections will increase their importance. The new methods allow for optimisation of inspection efforts with regard to cost/benefit ratio and offer synergies with nuclear security and counter-terrorism activities.

### 3.1. NDA and environmental sampling

In the past verification of nuclear material items by Non-Destructive Assay (NDA) was considered to serve safeguards and non-proliferation only. The minimum quantity of real concern was one Significant Quantity (SQ), amount of material sufficient to make a bomb out of. The imminence of nuclear terrorism has changed this view. One spent fuel item does not contain enough plutonium for a nuclear weapon. For terrorists, on the other hand, it would provide ample ingredients for a radiological dispersion device (RDD).

Since the end of 1980's Finland has had a systematic verification programme of nuclear material by NDA (Honkamaa and Hämäläinen, [1]). In 2006 the programme was formalised as part of the whole safeguards strategy. For the strategic planning period 2007-2011 we foresee two measurement campaigns at both Finnish nuclear power plant sites per year. The ultimate objective of the NDA programme is to serve the safeguards of final disposal: currently by producing continuously updated cumulative information on the contents of interim spent fuel storages, and eventually by partial defect testing of the spent fuel items to be subjected to final disposal. Additionally, as mentioned above, the NDA verification also contributes to security.

Environmental sampling as a tool for nuclear safeguards has unique potential to provide information on materials and activities (past or present) that enable assessing the correctness and completeness of declarations, on operator level and state level alike. In addition to the needs and objectives of the state safeguards system, a level of understanding of the environmental sampling and analysis techniques and their findings is essential in order to facilitate efficient implementation of the IAEA safeguards (Hämäläinen [2]).

### 3.2. Safeguards of final disposal of spent fuel

International specifications for safeguards of final disposal do not yet exist; hence safeguards currently applied are based on national requirements modified from the generic recommendations and outcome of the IAEA task "Safeguards for the Final Disposal of Spent Fuel in Geological Repositories". Our experiences are at the disposal of the international expert group set up to support the development of the IAEA safeguards for final disposal, whose successful future implementation in Finland, on the other hand, is in the vested interest of credible nuclear fuel cycle management.

Safeguards applied to the current Onkalo phase—excavation of the rock characterisation facility for the geological repository—follow the nature of the construction project, verifying excavated spaces against as-built-information. As complete design information of the final disposal facility is not available at this stage, detailed definition of the future safeguards

processes is not yet possible. However, based on the current plans, we have created a rough process model (Fritzell et al [3]), which can be used as a tool for further planning and identification of needs for research and development.

The disposal of nuclear fuel in a geological repository generates not only safeguards issues, but it incorporates also safety, security and social concern. Continuity of knowledge and security functions are to be maintained not only for an unlimited period of the validity Safeguards agreements but also to secure future generation in order to avoid unintentional or illegal intrusion to the repository. The role of security concern will began prominent already during operational time of the repository since the possibility to visual item counting or re-verification of the nuclear materials is lost during the encapsulation and emplacement process.

Finland and Sweden have strikingly similar final disposal concepts and the companies responsible for implementing them (Posiva in Finland and SKB in Sweden) cooperate closely with each other. Also the competent safeguards authorities (STUK in Finland and SKI in Sweden) intend to intensify their cooperation in the future, particularly in the areas of conceptual planning and research. The cooperation is naturally without prejudice to the independent position as competent authorities of each organisation in their respective countries.

## 4. Safeguards is security

In today's world nuclear security is more important than ever. The term is used here in its comprehensive meaning, encompassing nuclear and other radioactive materials and the measures taken to prevent, detect and respond to any illicit actions on the materials, their use, storage, disposal or transport (IAEA [4]). Additionally, we count in, preceding the three lines of defence (prevention, detection and response) the zeroth line, deterrence. While other authorities, such as law enforcement, bear the main responsibility for security as a whole, a state nuclear regulatory authority will inevitably have a role in nuclear security, counter-terrorism and radiological risk reduction.

Safeguards contribute to security, through established accountancy and reporting systems as well as the technical verification, containment and surveillance methods. A strong state safeguards system that consciously integrates security into its practices, combined with global international cooperation, will be effective also in fulfilment of the obligations set by international agreements such as the UN Security Council Resolution 1540. The responsibilities undertaken by States Signatories to the Resolution include measures to combat proliferation of sensitive materials through import and export controls and border monitoring, all of which are logically elements of both safeguards and security.

The agreements do not implement themselves, however: legislation must follow to oblige as well as empower the competent authorities in taking the responsibility of implementing those measures. We emphasise that this issue merits universal scrutiny, not limited, for example, to developing countries.

## 5. Cooperation

The threats to nuclear security and non-proliferation are inherently international, some of them global in scope. So should be the countermeasures.

Further enhancement of the cooperation in the area of border control and combating illicit trafficking within and without the European Union is of high importance. Uniformity and credibility of radiation detection at key transit nodal points would benefit from EU-wide implementation of the appropriate internationally agreed specifications and good practice. This is an area where Commission coordinated projects would have direct practical impact on security and non-proliferation safeguards.

Peer-to-peer support is an efficient way to assist in establishment and improvement of state systems for nuclear material safeguards, non-proliferation and security. The support may entail exchange of experiences and sharing of good practice, between nuclear security authorities as well as between nuclear operators. In the field of safeguards (not to mention safety) such support is well established; analogous development in the nuclear security sector is desirable. Existing safeguards cooperation fora could consider embracing security aspects.

There is a wealth of international conventions, agreements and initiatives intended to enhance the cooperation on nuclear security

and counter-terrorism. There are also some international guidelines, mainly those developed within the IAEA programme on nuclear security. What is largely missing is a comprehensive internationally agreed reference for a good state nuclear security system, including practical self-assessment guidelines. Such a reference would greatly assist in operationalising fully functional integrated state defence in depth concepts. We commend the IAEA endeavours underway to document the respective higher level guidance: objectives and fundamental principles as well as recommendations.

## 6. Conclusions

An effective national safeguards system shall enable the IAEA strengthened safeguards, implement the new Euratom safeguards approaches, and fulfil the obligations of the UNSC Resolution 1540. New methods for nuclear safeguards allow for optimisation of safeguards inspection efforts according to cost/benefit analysis and offer synergies with nuclear security and counter-terrorism coordination. The challenges are by no means over. In another 50 years' time safeguards shall be implemented for new types of facilities and activities, e.g. for final disposal of spent fuel: initial verification, surveillance during disposal process and containment within time frames unique to any previous safeguards application. Security-related activities are likely to have developed considerably. Integrated comprehensive defence in depth concepts encompass the common objectives and complementary methods of safeguards and security. We suggest that these concepts are best operationalised by taking advantage of international cooperation. Here the international organisations and inspectorates—Euratom safeguards and IAEA safeguards and security—have an opportunity to provide added value to member states in a coordinative role.

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# Session 14

# Remote sensing

# CSSP Investigations using SAR Imagery for Safeguards Applications

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***Abstract:***

*The Canadian Safeguards Support Program (CSSP) of the Canadian Nuclear Safety Commission has been investigating potential applications of SAR imagery for nuclear safeguards during the last several years. Particular strengths of synthetic aperture radar (SAR) satellite imagery are highlighted and potential applications of this all-weather, day/night information source to safeguards are described and illustrated by examples. Our investigations cover several areas: object detection, infrastructure analysis, surface change detection, and interferometry. Airborne and satellite imagery of various facilities including nuclear sites have been used in the analyses with particular emphasis on the detection of clandestine operations.*

*A summary of the work done to date will be discussed along with details of a new project which is jointly supported by various government departments in Canada. High resolution SAR imagery from new satellites such as RADARSAT-2 and TerraSAR-X may be available in the near future to be included in our initial phase of the project.*

**Keywords:** SAR satellite imagery, safeguards application, SAR image analysis

## 1. Introduction

Satellite imagery has become an important tool for the International Atomic Energy Agency's (IAEA) nuclear safeguards verification. In support of the IAEA in this activity, the Canadian Safeguards Support Program (CSSP) has carried out a comprehensive program that includes the entire spectrum of satellite imagery ranging from the optical panchromatic, multispectral and hyperspectral imagery to imagery obtained with active sensors such as Synthetic Aperture Radar (SAR).

The main objective of our program is to investigate the utility of these remote sensing tools and to develop methodologies and analysis techniques that could be used in nuclear safeguards verification, particularly in the area of detection of clandestine activities.

In collaboration with other national and international organizations and engagement of consultancy services from private companies the CSSP has begun its investigation of high resolution SAR obtainable with satellites such as the Canadian RADARSAT-2 and the European TerraSAR-X which are scheduled to be launched in 2007.

This paper will discuss several areas where SAR imagery could be used for object detection, infrastructure analysis, change detection which includes the use of colour composite, coherent change detection and interferometry.

## 2. Object Detection

SAR offers a different picture of the world than optical images. When monitoring areas for the purpose of nuclear safeguards, a consistent method of acquiring data is important. Optical imaging satellites

often have higher resolution than their radar counterparts, and optical images are often much easier to interpret visually by inexperienced users. Despite these factors, SAR has particular strengths for the purpose of a reliable imaging system.

SAR is an active system, meaning that it does not rely on sunlight for capturing images. Unlike optical systems, SAR provides its own illumination and is able to take images at any time of the day and night. This is particularly advantageous for imaging in the far north, when there is only a few hours or less of daylight during the winter. Furthermore, SAR operates in the microwave band of the electromagnetic spectrum, and as such is able to see through clouds, fog, rain, and weather conditions such as hurricanes. Many tropical and costal regions are almost always covered by clouds making it virtually impossible to get optical images.

Figure 1 shows a RADARSAT-1 image of Rauma, Finland acquired on November 17, 2005. The channel markers leading into the port area are easily identified next to the calm water. The town of Rauma is also quite visible on the right side of the image as bright structures.

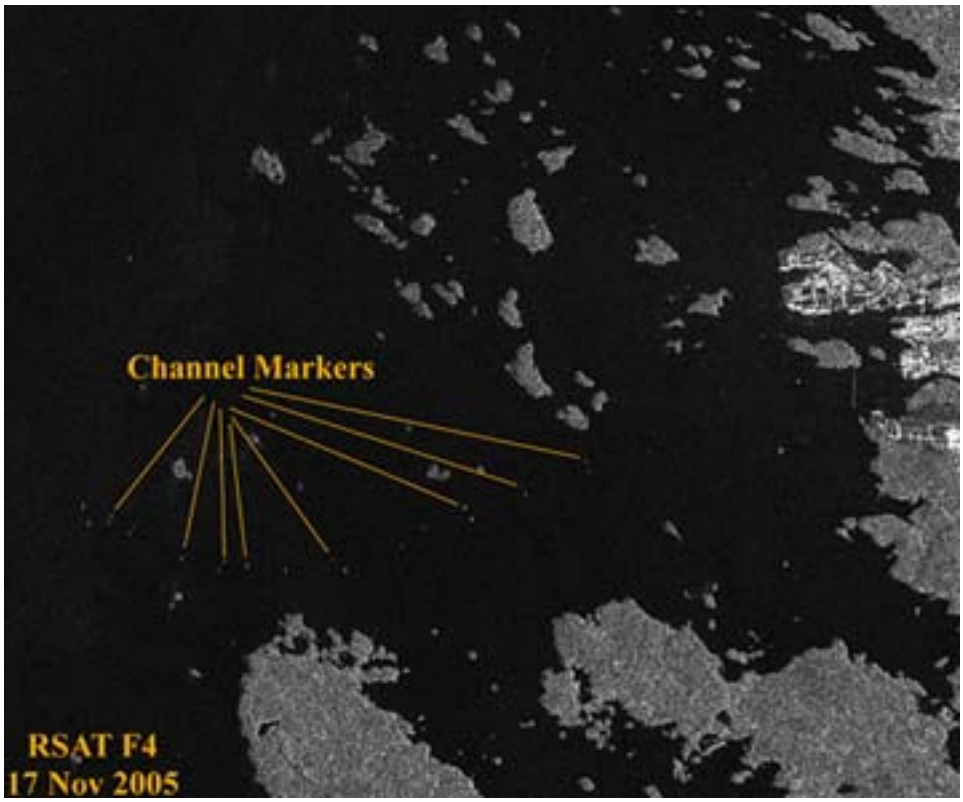


**Figure 1. RADARSAT-1 image of channel markers near Rauma, Finland, approximately 13 km south of Olkiluoto nuclear plant. The channel markers are quite visible against the calm water.**

Figure 2 shows the same location acquired four months later, on March 17, 2006, when the water areas are frozen over with ice. The harbour ice is noticeable when compared to the dark water of Figure 1. The ship trails through the ice are clearly visible due to the rough nature of the broken ice in the trails. Also noted is an area of open water where the trail seems to disappear.

SAR measures the backscattered response from an area on the ground. Smooth, flat areas such as calm water, roads and runways will typically scatter much of the transmitted signal away from the radar and show up as dark areas in the radar image. Buildings tend to bounce much of the signal back towards the radar and are represented as bright areas in the image. This illustrates one of the strengths of SAR, the ability to locate many types of man made objects such as buildings, roads, bridges and fences.

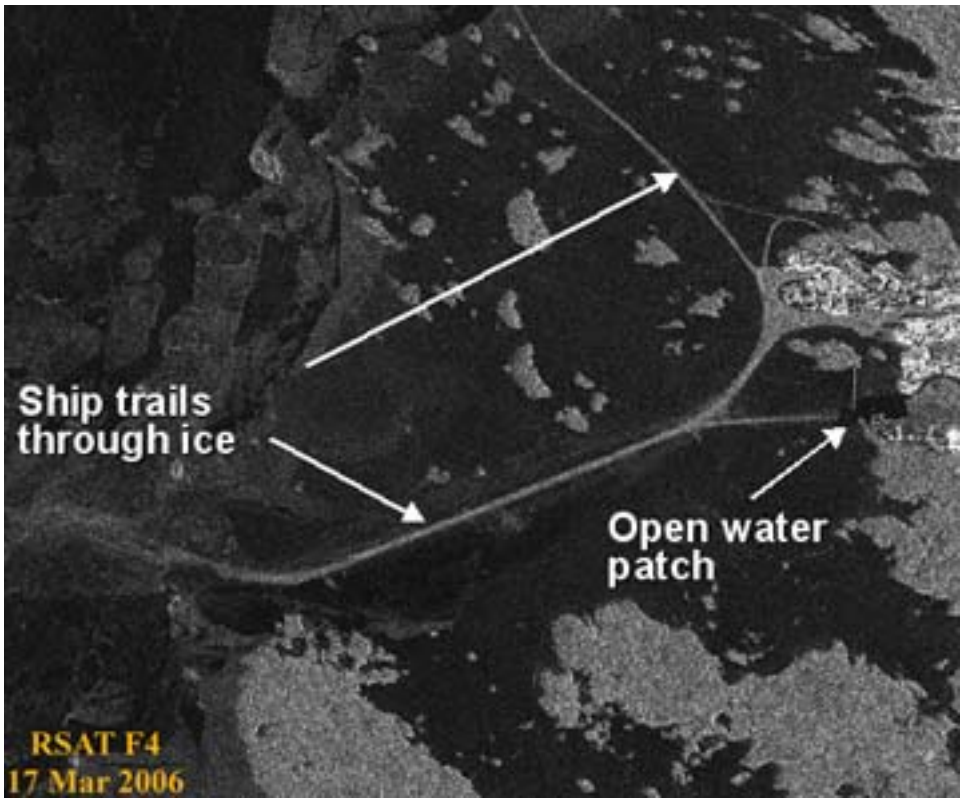


**Figure 2. RADARSAT-1 image of the same area as Figure 1 showing ship trails through ice. The trails show up brighter than the surrounding ice because of the rough nature of the disturbed trails. Also indicated is an area of open water where the trail disappears.**

## 3. Infrastructure Analysis

Analysing images for different types of infrastructure is another important task for nuclear safeguards. The ability to remotely identify potential nuclear sites based on the type of buildings, pipes, and power lines can be greatly enhanced by using Polarimetric SAR (PolSAR) instead of conventional SAR. PolSAR uses the fact that the response from ground targets depends on the polarization of the imaging signal. It has been well documented [1,2,3] that different shapes on the ground will show up better at different polarizations. By transmitting and receiving radar signals at multiple polarizations, PolSAR can reveal more information about the ground structures. In contrast, conventional SAR only transmits and acquires signals at one fixed polarization and might not detect certain shapes, while PolSAR data can be processed to create ground images at every possible polarization. This ability allows a user to study different views of the one polarimetric image in which different structures will come in and out of 'focus'.

Figure 3 shows images at three different polarizations as well as a composite image made from the three polarizations of the Gentilly nuclear site in Quebec, Canada. In the top right image the fences and barriers are easily detected compared to the top left image, although the point objects clearly visible in the top left image are not as apparent.

PolSAR actually acquires four images for one satellite pass. These four images are the primary polarization states and are described as HH, HV, VH, and VV. HV means that the SAR transmits a horizontally polarized signal and receives only in the vertical polarization. Any other polarization angle can be calculated from these four polarizations in a process called polarization synthesis.

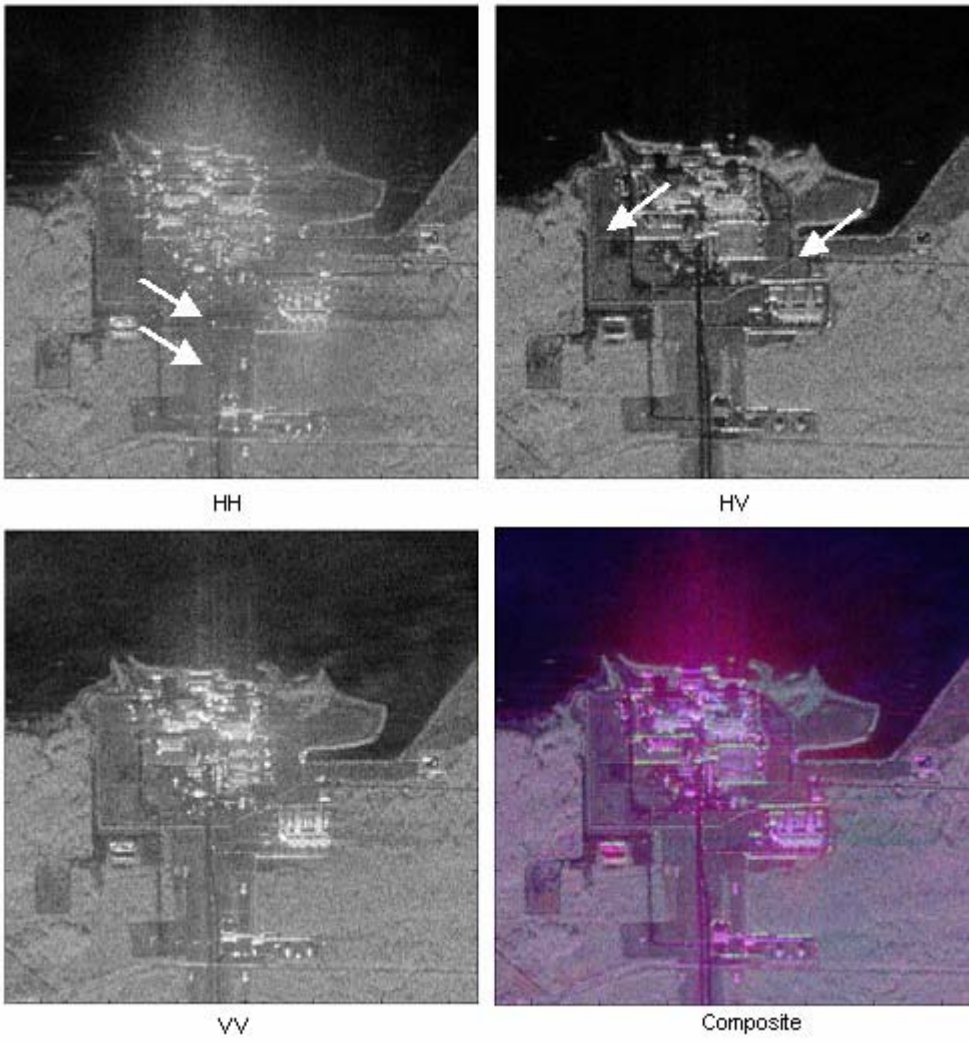


**Figure 3. CV-580 Airborne polarimetric SAR (PolSAR) images of Gentilly nuclear plant in Quebec, Canada. Three polarizations (HH,HV,VV) along with a composite of the three images are shown. The VH polarization is very similar to the HV polarization and is not shown. Arrows in the top images indicate points (left) and fences (right).**

## 4. Change Detection

There are many inherent problems when using satellites to remotely monitor areas for normal or clandestine activities. It is very difficult to capture an activity with a satellite, as the timing of the satellite passing overhead has to match the activity occurring, not even considering that the lighting and weather is also a factor for optical systems. With an understanding of satellite orbits, a person could determine when the satellites are passing overhead and thus avoid being imaged.

These problems, however, can be overcome with change detection techniques. Only the most simple of these techniques is available to optical imagery, the remainder are specific to radar and require additional processing.

### 4.1. Colour Composite

A colour composite image is an easy and simple way to show the changes between three images taken at different times. Each image is given a distinct colour, red, green or blue, and the three images are combined.

Figure 4 is a colour composite of RADARSAT-1 images showing the Bullfrog mine near Las Vegas, Nevada. The three images were acquired on October 1, 2000, November 18, 2000, and October 4, 2004. The black, grey and white shades indicate areas that have not significantly changed across the three acquisition dates. The colours in the composite image indicate areas where there is a difference in the backscattered radiation of certain dates. For example, the yellow areas in Figure 4 indicate features that are only found in the October 1, 2000 and November 18, 2000 images. The other colour combinations are shown in the inset colour map.

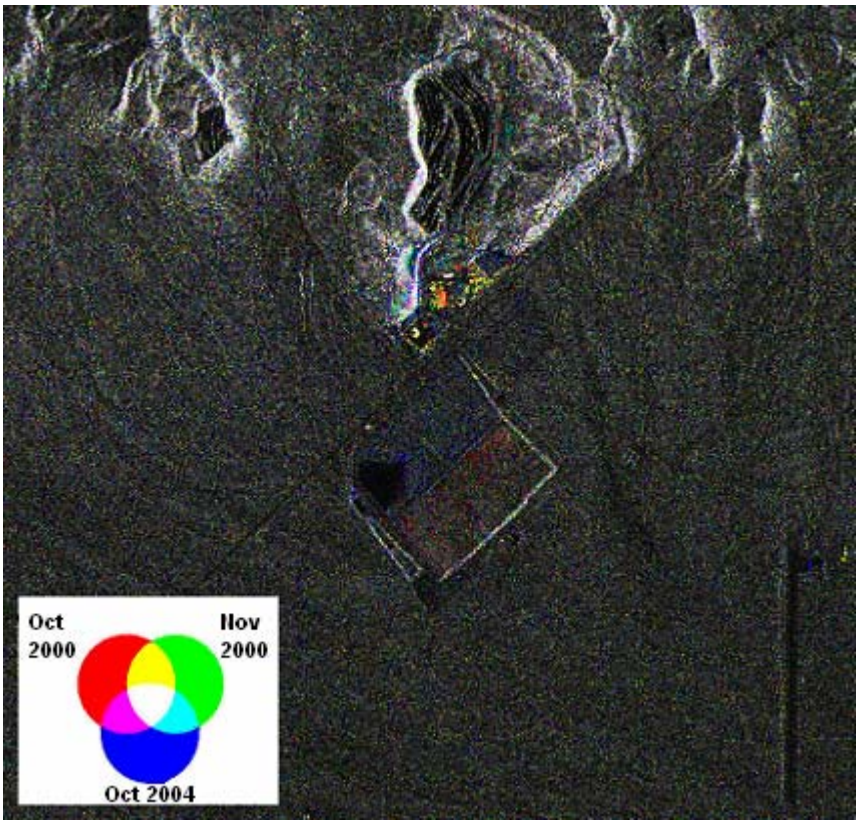


**Figure 4. Colour composite showing the Bullfrog mine in Nevada, USA. The three RADARSAT-1 images collected on October 1, 2000, November 18, 2000, and October 4, 2004 are shown in red, green, and blue, respectively.**

## 4.2. Coherent Change Detection

SAR also offers the benefit of detecting activities that are a particular concern for nuclear sites, both domestic and foreign. Through a technique called Coherent Change Detection (CCD), subtle changes on the ground can be identified between two SAR images. This is of obvious benefit for monitoring the condition of security fences or concrete barriers, or for monitoring mining activities. Although these types of actions could be identified in conventional optical and SAR imagery, the CCD is sensitive enough to detect if the object had been disturbed between the satellite passes. For example, if a section of tailings pile from a uranium mine has been dug up and then refilled between satellite passes, conventional imaging would not notice the activity. CCD is able to detect the disturbance because it is impossible to replace the tailings in the exact location as before. This is also true if a concrete barrier has been removed and then replaced; it will never be in the exact location again. The subtle surface change of a vehicle driving repeatedly on a dirt road is observable with CCD. In fact, if the resolution of the SAR is sufficient, it is even possible to identify footprints [4]. CCD does have limitations in that it is very sensitive to ground moisture and the movement of vegetation.

Figure 5 shows a CCD image (right) of the Bullfrog mine and the two RADARSAT-1 images it was produced from (left and center), acquired on October 1, 2000 and November 18, 2000, respectively. The dark areas indicated by the arrows represent that there has been some disturbance of the ground.

Examining records of the mining activities confirms that there was re-sloping of the waste rock piles in these locations during the two satellite passes.

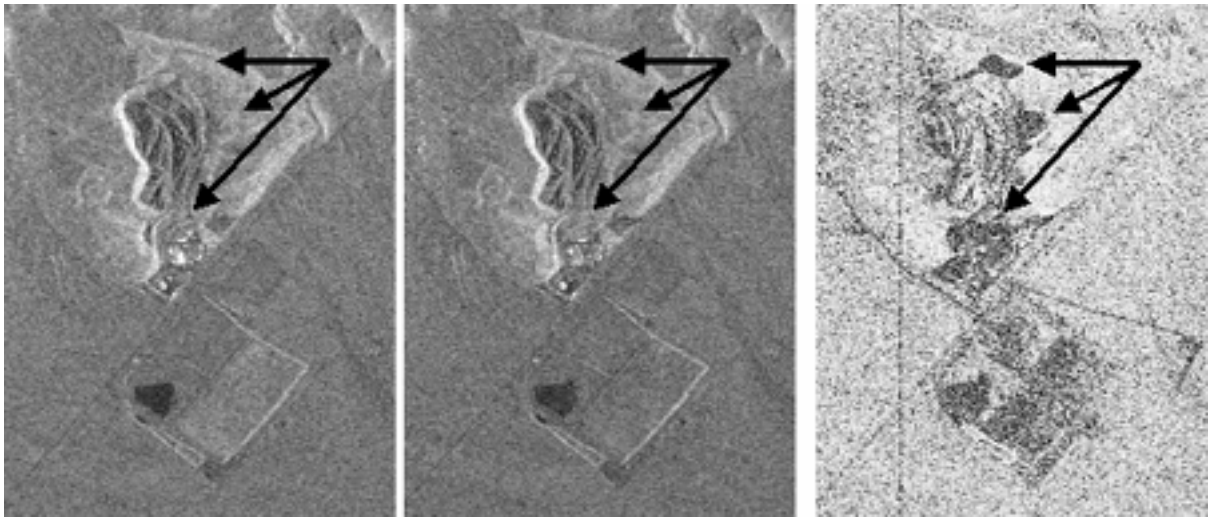

**Figure 5. CCD image (right) of the Bullfrog mine in near Las Vegas, Nevada. The CCD image was produced from the two RADARSAT-1 images (left and center) from Oct. 1, 2000 and Nov. 18, 2000 respectively. Note that the areas indicated in the CCD image do not show up in the conventional images.**

### 4.3. Interferometry

CCD is a method for identifying that there has been a change in a certain area, but it is often important to get a quantitative measure of how much an area is moving, or subsiding. This can be achieved by Interferometric SAR (InSAR), and can be used to detect millimetre-sized movement in the scene. InSAR can detect subsidence caused by underground activities such as subway tunnels [5], attempting to tunnel to circumvent surface security features, and the effect of underground mining activities. Again, this could be of particular importance for monitoring uranium mines for signs of activity.

As with CCD, InSAR is sensitive to moisture and the movement of vegetation, so two variations have been developed. Differential InSAR (DInSAR) is used in dry, arid conditions where most of the ground structures are stable over long periods of time, and Persistent Scatterer InSAR (PSInSAR) is used for vegetated and urban environments where only a limited number of structures are stable enough to provide subsidence information.

An example of DInSAR is illustrated in Figure 6, which shows the Nevada Test Site. The interferogram was overlaid on an optical IKONOS image of the same area. The areas of greatest subsidence are shown by the red areas and are located near the sites of the underground nuclear tests.

PSInSAR identifies specific points in the scenes that are not changed by the moisture and vegetation problems. Since these points are usually the result of man-made objects, such as fences, poles, and features on buildings, PSInSAR is particularly suited for urban environments. A consequence of using isolated persistent points is that many more scenes are necessary. For the case shown in Figure 7, from Dixon et al. [6], 33 RADARSAT-1 images taken from 2002 to 2005 were used. The rates of subsidence are shown for the identified points in New Orleans, Louisiana. Particularly high rates of subsidence can be identified near the MRGO canal, which was breached by Hurricane Katrina in August 2005.

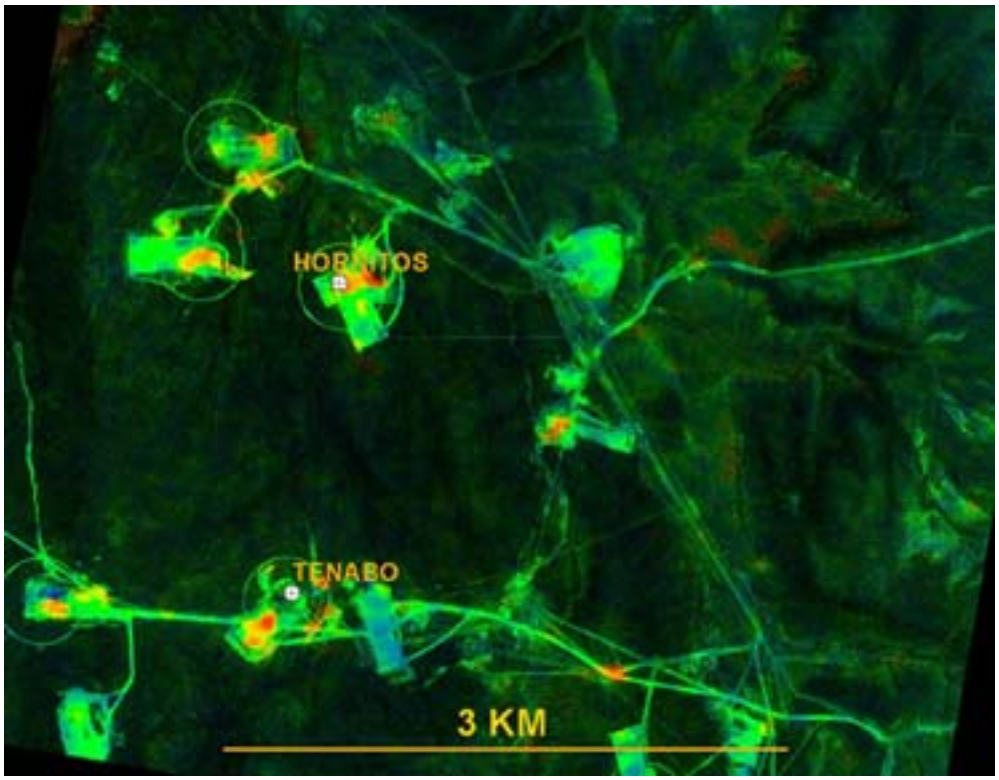


**Figure 6. Interferogram overlaid on an optical IKONOS image of the Nevada nuclear test site. The interferogram was created from RADARSAT-1 images taken on Oct. 1, 2000 and Nov. 18, 2000. The red areas indicate the regions of greatest subsidence and are found at underground nuclear test sites.**

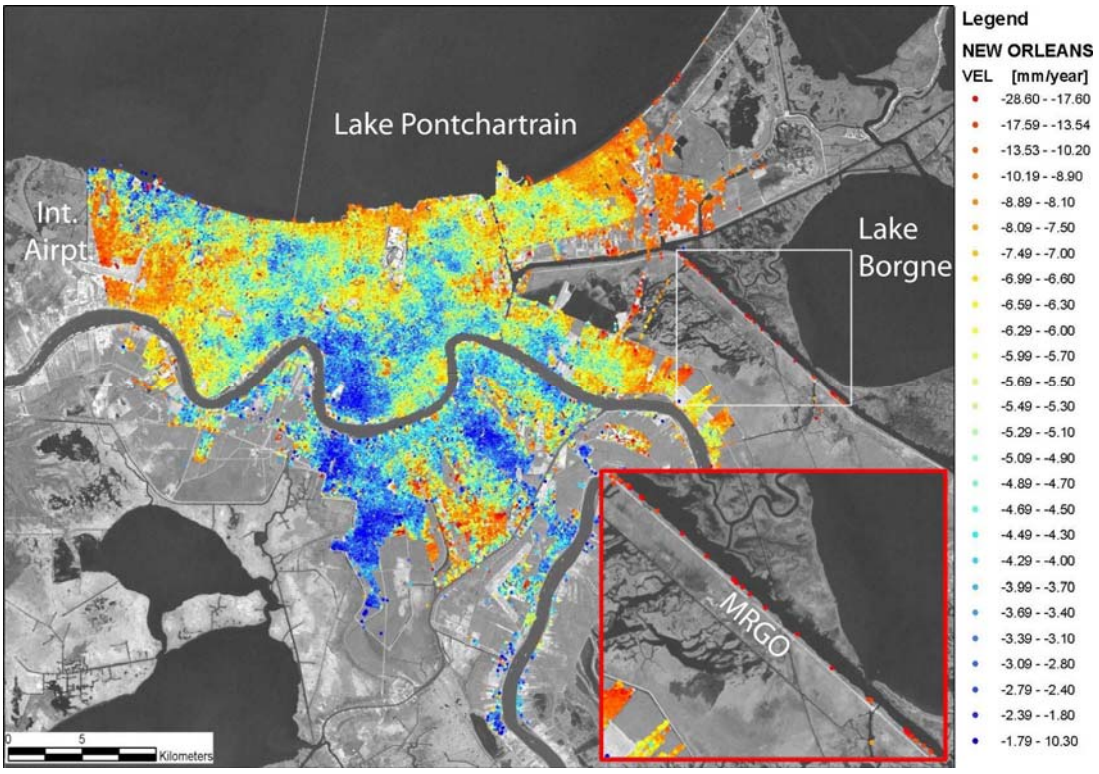


**Figure 7. Persistent Scatterer locations and rate of subsidence in mm/year in New Orleans, Louisiana. 33 RADARSAT-1 scenes from 2002-2005 were used. The MRGO canal shown in the inset was breached during Hurricane Katrina and has high levels of subsidence. From Dixon et al.[6].**

## 5. Conclusion

The particular strengths of SAR imagery for nuclear safeguards applications have been outlined. SAR is a 24-hour, 7 days a week, all weather system that can be used to monitor sensitive areas safely. In addition to producing visual images that offer a different perspective than optical images, there are advanced processing techniques that are specific to radar systems that allow a qualitative and quantitative measure of surface disturbances.

All of the techniques presented are affected by the resolution of the SAR. As the technology progresses and additional satellites with increased resolution are created, the benefit of these techniques will also increase.

SAR and optical imaging systems should not be treated as two competing alternatives for remote sensing, but rather as two methods with their own strengths and weaknesses that can be used to complement each other for nuclear safeguard applications.

## 6. Acknowledgements

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- Defence Research and Development Canada – Ottawa (DRDC-O): Karim Mattar, Chen Liu, Paris Vachon, and Gary Geling
- The Canadian Space Agency (CSA): Christine Giguère.

## 7. Legal matters

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# Practical training in hyperspectral satellite image analysis

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***Abstract:***

*Hyperspectral remote sensing holds the promise of providing inspection agencies such as the IAEA the capability of chemical analysis, at standoff distances, of rocks, ores and other stockpiles that are impossible to differentiate with high-resolution panchromatic satellite imagery. The Canadian Safeguards Support Program (CSSP) has been investigating hyperspectral methods and applications for nuclear safeguards since 2001. In 2004, we began a series of practical, hands-on training workshops in hyperspectral image analysis for image analysts. Rather than provide an intensive academic treatment of the subject available elsewhere, these workshops provide a very short, focussed introduction to hyperspectral satellite imagery of most relevance to the Agency.*

*The workshops assume limited advance knowledge of hyperspectral science, but some basic experience with ENVI™, a commercial image processing software package commonly used for hyperspectral analysis. After a brief discussion of hyperspectral remote sensing, the characteristics of the American HYPERION satellite currently providing hyperspectral imagery, and an overview of hyperspectral methods, participants are led through data preparation (removal of instrument artefacts) and atmospheric correction, prior to actual image analysis. A proprietary pre-processing software 'Wizard' insert for ENVI™ written for the CSSP is provided to automate most of the data preparation. After a discussion of matching analysis tools and techniques with the question at hand, selected ENVI™ image analysis tools are presented. Participants use standard ENVI™ tools and the provided Wizard to analyze actual HYPERION imagery of a safeguards relevant site. A training manual with step-by-step instructions is supplied to the participants for future reference.*

*Our workshops have been successful in giving participants some practical experience, and providing enough experience, confidence and resource material with which to begin to perform hyperspectral analyses in support of inspection activities.*

**Keywords:** remote sensing; hyperspectral; satellite; imagery; training; analysis

## 1. Introduction

Since 2000 the IAEA have applied satellite image analysis in their nuclear safeguards monitoring program. Specialized image analysts primarily interpret high spatial resolution imagery from sensors such as Quickbird or IKONOS for signs of activity at nuclear facilities and mines, based on visual cues such as vehicle tracks, configuration and changes in structures including buildings, ore and rock piles, and roads. Other sensors such as LANDSAT or ASTER also offer thermal information that is used to detect features such as warm water discharges from nuclear plants.

With the launch of an Earth Observation satellite EO-1 HYPERION in late 2000, a third class of satellite-based image data became available. The hyperspectral capabilities of HYPERION offer detailed spectral information in the visible and shortwave infrared (wavelengths in the range 400-2500 nm). The strength of hyperspectral is its potential for remote chemical characterization of rocks, ores and other stockpiles that would otherwise be impossible to differentiate using standard spatial interpretive methods.

In 2001 the Canadian Safeguards Support Program (CSSP) began to investigate the utility of hyperspectral in safeguards applications, and engaged Borstad Associates Ltd. to conduct a series of case study analyses. In 2004 a hyperspectral training workshop was offered for the first time to the IAEA Satellite Image Analysis Unit (SIAU) analysts, and in response to their feedback the workshop was repeated in 2005 and 2006, with updates each year to include additional topics as well as new developments in data processing.

## 2. Course development

By 2003 a number of courses were already being offered by universities and other educational institutions on hyperspectral imaging and image interpretation, and in fact members of the SIAU had participated in such workshop before receiving our training. This prior workshop provided a good theoretical background, but there was interest in additional focussed, hands-on training. Our course was therefore structured around the processing and analysis of EO-1 HYPERION imagery, since this is currently the only available open source of satellite-based hyperspectral imagery. Software and training datasets were developed directly from the safeguards case studies performed in the preceding two years, and so were of direct relevance to the participants. Techniques introduced during the course also focussed on those most applicable to safeguards analyses. It should be noted that the same processing and analysis techniques discussed here are applicable also to hyperspectral imagery obtained from airborne platforms or hand-held devices.

The 2004 course was considered by participants to be highly successful and a “refresher” course was held the next year. Since the majority of 2005 attendees were repeat participants from 2004, the introductory portion of the course was minimized, and some new and updated techniques were included, including a customized image pre-processing software “wizard” that sped up much of the tedious process of image correction required to make the imagery ready for analysis. In addition, a step-by-step guide was distributed to participants to assist with analyses undertaken after completion of the workshop. By the end of the 2005 workshop participants had performed all of the processing steps and analytical techniques then in routine use at Borstad Associates, and in the time between the 2005 and 2006 workshops, SIAU analysts performed successful hyperspectral analyses as part of their regular duties.

In 2006 all of the workshop participants were new or recent additions to the SIAU and so had varying degrees of previous experience with image analysis. The workshop was therefore restructured to include more theoretical background than the previous two years, though due to increasingly streamlined preprocessing, there was sufficient time to introduce all of the analytical techniques from the 2005 workshop. In addition, a new module was presented that allows analysts to visualize both hyperspectral imagery (or derived maps) and the more traditional high spatial resolution imagery as overlays, so bridging the two technologies and enabling analysts to benefit from their simultaneous interpretation.

## 3. Setup and Software

The workshops were structured as hands-on sessions, with each participant or pair of participants at a workstation. Following brief explanatory lectures, participants were guided through test datasets to gain experience performing each preprocessing and analytical step.

Image processing and analysis software was based on ITT ENVI version 3.2 in 2004, progressing to version 4.2 and 4.3 in 2006 as new versions were released. Although other commercial image processing software is available, ENVI is well suited to hyperspectral and is in common use in the remote sensing community. The workshops and step-by-step guide were not intended as ENVI training, but identified those tools most useful for hyperspectral analysis, and most importantly assisted participants in the selection of analytical algorithms appropriate to the scenarios likely to be encountered in safeguards-related analyses. New features available with updated versions of ENVI were introduced in the 2005 and 2006 workshops as appropriate, including new atmospheric correction algorithms and multi-resolution overlays as described above. As well, participants were alerted to potential problem areas and unresolved software “bugs” encountered during our own experience. ENVI software was supplemented with custom routines developed by Borstad Associates

Ltd, notably the preprocessing software wizard that automated much of the initial correction required with HYPERION imagery.

## 4. Course content

For analysts experienced in working with panchromatic or multispectral imagery, the concepts and interpretive techniques employed in spectral analysis are often unfamiliar, so the introductory session to the course in all three years reviewed the nature of hyperspectral imagery and analysis. A sample explanatory slide is illustrated in Figure 1.

The second workshop session in each year was devoted to image artefact correction. Unlike panchromatic or multispectral, hyperspectral satellite technology is still relatively new, and so the at-source preprocessing tends to be incomplete, and the imagery provided by the supplier requires considerable preparation to remove small but important instrument errors. In the first year of the workshop, artefact removal was performed “manually” – that is, using a collection of standard image processing tools available with ENVI, but in 2005 and 2006 much of this time-consuming process was accomplished using a custom software wizard designed specifically to remove artefacts from HYPERION imagery and to format it for ENVI analysis.

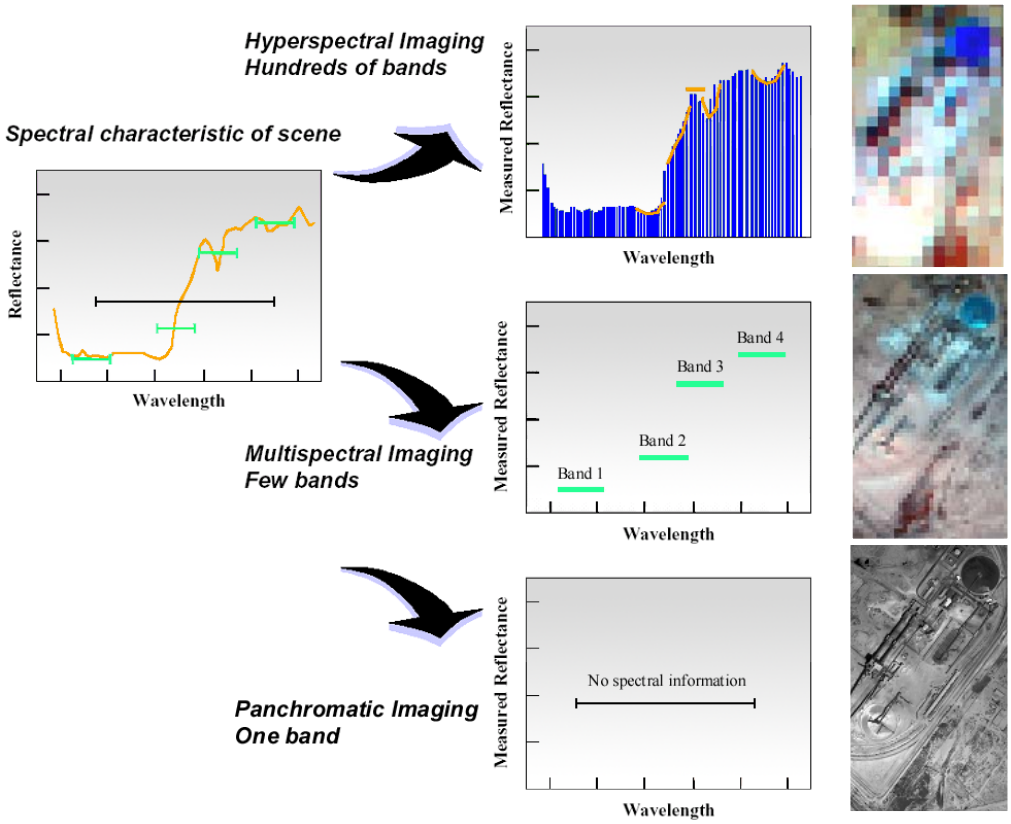


**Figure 1:** Introduction to hyperspectral imaging

The third workshop session covered atmospheric correction. Because hyperspectral analysis frequently involves the identification of unknown materials present in an image, the image spectra must be comparable to reference spectra of known materials from field or lab measurements, and hence must be corrected to remove the very large spectral signal due to the earth’s atmosphere. Figure 2 illustrates the magnitude of the correction. Workshop participants were introduced to and obtained practical experience with both theoretical (model-based) and empirical methods of atmospheric correction. Decision-making as to the need for atmospheric correction on a case-by-case basis was also discussed.

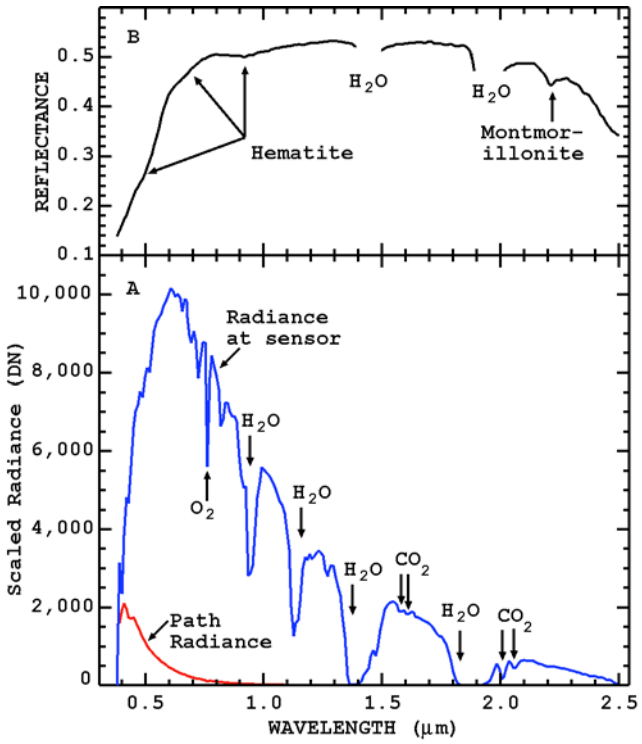


**Figure 2:** Comparison of uncorrected (lower) and atmospherically corrected (upper) spectra for the same material (*speclab.cr.usgs.gov/PAPERS.calibration.tutorial/).*

| **If your question is...** | **The recommended algorithm to use is...** |
|---|---|
| Is the material at location *x* found elsewhere in the scene? Or, is material "A" found anywhere in the scene? | Supervised classification |
| What are the relationships among this user-defined set of materials found in the scene? | Spectral Analyst |
| What are the relationships among all of the pixels in the scene? | Unsupervised classification |

**Table 1:** Selection of analytical algorithms.

About half of the overall workshop schedule was devoted to the selection and application of analytical techniques. The three most useful techniques are listed in Table 1, along with typical safeguards situations in which each would be used. A case study for each was presented, and participants were guided through the analyses during the workshop. Figure 3 illustrates one of these case studies in which phosphate ore and the refined fertilizer product was mapped at Al Qaim, Iraq. Lab spectra for these two materials were used to locate them in a HYPERION scene over Al Qaim. In 2004 this analysis was performed "manually" using individual tools available in ENVI 3.2. In 2005 and 2006, using more recent ENVI 4.1 and 4.2 participants were introduced to the "Spectral Hourglass Wizard" that automates much of the analysis, thereby reducing overall processing times. Figure 4 illustrates the application of ENVI's Spectral Analyst to determine the similarities among materials in a HYPERION scene without necessarily identifying them. In this exercise, the operational relationships between excavations and rock and ore piles at Ranger Mine, Australia were inferred from their

spectral similarities. The rock from three different excavation sites at the same pit was traced to various locations around the mine.

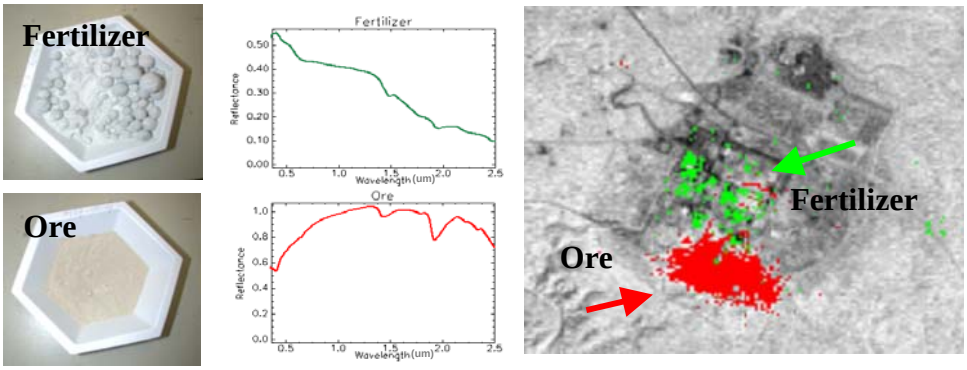


**Figure 3:** Case study of supervised classification at Al Qaim, Iraq.

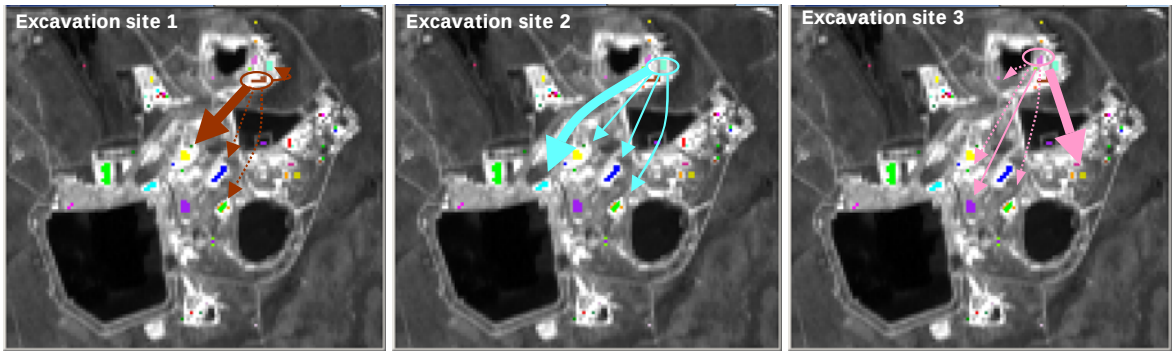


**Figure 4:** Spectral Analyst case study at Ranger Mine, Australia.

In 2006, new techniques were also introduced, showing SIAU analysts how the geochemistry determined from hyperspectral data can be interpreted in light of high-resolution panchromatic imagery of the same area, or conversely, how activities observed from high resolution imagery can be interpreted in terms of the geochemistry.

## 5. Course feedback

The final session in each year was a summary and evaluation session in which participants were tested on their understanding of the material and the potential safeguards applications of the technology, as well as providing feedback on the workshop and suggestions for future training. The feedback from participants in each year was extremely positive. The interactive format with hands-on practice was appreciated, as was the balanced presentation showing both the strengths of the technology and potential for error. The perceived level of difficulty of the material varied with the background experience of the participants, but the overall level of understanding was high as was the recognition of potential safeguards applications. The compilation of a written guide in 2005 was well received, both during the workshop and as a future reference. Participants who attended in both 2004 and 2005 appreciated the timesavings achieved with the introduction of processing software wizards.

Participants also demonstrated their understanding by pinpointing areas for potential technological improvement that are currently recognized within the remote sensing community, including the need for improved atmospheric correction algorithms and for specialized spectral libraries more relevant to safeguards applications. All requested ongoing training in the form of repeat or annual refresher workshops for new and existing staff.

## 6. Conclusion

Our workshops have been successful in providing participants with enough practical experience, confidence and resource material with which to begin to perform hyperspectral analyses in support of inspection activities. The annual format permits analysts to be kept informed of new developments in this young technology. The interactive nature of the workshops and feedback from participants has enabled us over the 3 years to respond to their needs by incorporating new workshop modules and by developing automated software routines to improve processing times.

HYPERION is now in extended mission, having fulfilled its original goal as a technology demonstration. Several countries are now planning new hyperspectral sensors with higher resolution and better signal-to-noise, scheduled for launch in the next 3 to 5 years. With these new sensors, hyperspectral remote sensing is expected to make the transition from research to operational mode, significantly increasing data coverage and availability. Providing training for agencies such as the IAEA will ensure that they are fully prepared to take advantage of this emerging technology.

# Thermal Infrared Satellite Imagery Analysis for Nuclear Safeguards Applications

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***Abstract:***

*The use of satellite imagery for nuclear safeguards applications today is very much limited to visible and near infrared data, due to at least two reasons: First, from a technical point of view, these data provides the best spatial resolution in the sub-meter range for the monitoring of small-scale nuclear-related activities. Second, from the user's point, the (visual) interpretation of optical data is more obvious rather than the analysis of thermal infrared, hyperspectral or radar image data where extensive pre-processing and knowledge on the sensor is required. However, also satellite data from thermal infrared, hyperspectral and microwave sensors involves information being relevant for nuclear monitoring.*

*The application of thermal infrared imagery for the monitoring of (heat generating) nuclear activities seems to be reasonable, even though the spatial resolution of satellite-based thermal infrared sensor bands is still limited to 60m (LANDSAT 7) and 90m (ASTER) respectively. The image data given by the thermal infrared system enables the user to analyse thermal differences between the area of interest and its neighbourhood and thus to derive information on the operational status of the facility.*

*The given paper proposes two easy-to-use approaches for the advanced analysis of thermal infrared satellite imagery. The first technique represents the application of a wavelet-based spatial sharpening to thermal infrared band. Second, the so-called Reed-Xiaoli Detector (RXD) algorithm was tested for detecting the spectral or colour differences between a region and its neighbouring pixels or the entire data set. Both procedures are implemented in the ENVI system.*

**Keywords:** thermal infrared satellite imagery; data fusion; anomaly detection; sub-pixel analysis

## 1. Introduction

The use of satellite imagery for nuclear safeguards applications today is very much limited to optical data, due to at least two reasons: First, from a technical point of view, optical data provides the best spatial resolution in the sub-meter range for the monitoring of small-scale nuclear-related activities and structures. Second, from the image analyst's point, the (visual) interpretation of optical data is more obvious rather than analysing thermal infrared, hyperspectral or radar image data where extensive pre-processing and knowledge on the sensor is required. However, also satellite data from thermal infrared, hyperspectral and microwave sensors involves useful or even relevant information for nuclear monitoring, which probable does not exist in the optical data.

Thermal infrared sensors measure the thermal radiation from the earth surface. As thermal infrared energy is emitted from all objects having a temperature above absolute zero, man-made and natural surfaces emit thermal infrared electromagnetic radiation on a typical day, partly during the night. Thermal infrared data may therefore be a good candidate for monitoring heat emissions and thermal

anomalies at nuclear facilities, and to assess the operational status of the sites. More information on thermal infrared remote sensing is given in [1-4].

The resolution of today's thermal infrared satellite sensors, 60m for LANDSAT 7 and 90m for ASTER, doesn't show many promises for nuclear monitoring. However, image processing techniques may provide useful tools for the visualisation of thermal information. Two ease-to-use procedures will be presented in this paper. The first technique, developed by Ranchin and Wald [5], represents the application of a wavelet-based spatial sharpening to thermal infrared bands. Second, the so-called Reed-Xiaoli Detector (RXD) algorithm by Chang and Chiang [6] was tested for detecting the spectral or colour differences between a region and its neighbouring pixels or the entire data set.

## 2. Thermal Infrared Data Availability from Space

The only thermal infrared sensors with a commercial payload are given by the Landsat-5, Landsat-7 and ASTER satellites. Due to the longer wavelengths in this part of the electromagnetic spectrum, data obtained with thermal infrared sensors have as a general rule much lower spatial resolutions compared to optical, near infrared and short-wave infrared sensors. The LANDSAT project is supported by U.S. Geological Survey (USGS) and the National Aeronautics and Space Administration (NASA). The Advanced Spaceborne Thermal Emission and Reflection Radiometer (ASTER) is an advanced multispectral imager that was launched on board NASA's Terra spacecraft in 1999.

LANDSAT 7 ETM+, launched in 1999, holds one thermal channel in the range of 10.40 to 12.50 µm with a spatial resolution of 60 m and a temperature accuracy of 0.5°K. ASTER features the thermal subsystem TIR with five channels between 8.125 and 11.65 µm at a spatial resolution of 90m and a temperature accuracy of 0.3°K. Please see Figure 1 for a comparison of the spectral and spatial resolution of ASTER and LANDSAT-7; more information on these sensors is given in Kramer [7].

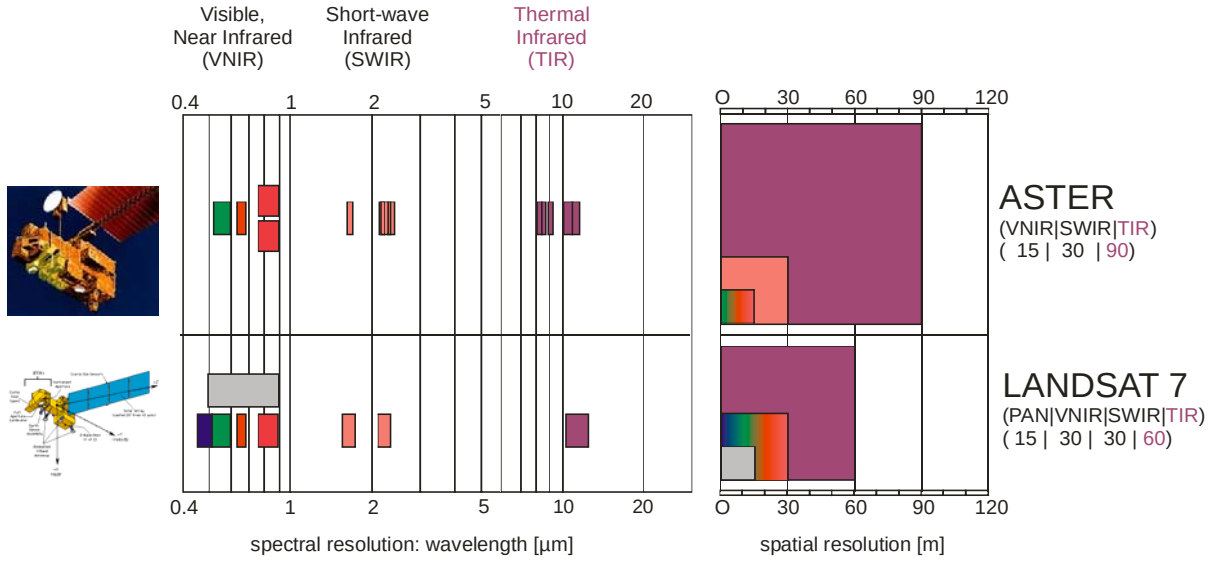



**Figure 1:** Comparison of the spectral and spatial resolution of ASTER and Landsat-7 ETM+

## 3. Temperature Estimation

Thermal infrared energy is emitted from all objects having a temperature above absolute zero. As a result, man-made and natural surfaces emit thermal infrared electromagnetic radiation on a typical day, partly during the night. Since different materials (soils, plants, water, man-made materials) selectively absorb short-wave solar energy and radiate the long-wave (thermal) energy in a specific way, it is possible to determine the type of material based on the thermal emission characteristics of the material and to evaluate whether significant changes have taken place in the thermal characteristics of these materials over time.

### 3.1 Data Correction and Enhancement

Multispectral TIR data requires a special approach for data processing. First of all, the data needs to be calibrated. During scanning, the scanner sensors the blackbody, and these data are used for calibration. Then, the at-sensor radiance needs to be corrected for atmospheric effects. This correction results in radiant temperature images. After that, the data are processed to derive spectral emissivity and surface kinetic temperature values; a number if approaches exists.

A simple approach is the normalized emissivity approach. In this method, for a particular pixel, the highest of the set of temperatures in various channels is considered as the near-correct temperature of the pixel. The highest temperature and a fixed spectral emissivity value of 0.96 are used to compute the surface (kinetic) temperature. From the kinetic temperature, spectral emissivity is computed using the corresponding radiant temperatures. In this way, spectral emissivity images are derived for various channels. The various spectral emissivity images, and also the radiant temperature in general, are highly correlated. The contrast ratio in spectral emissivity is commonly less than 0.15, as compared to the contrast ratio of 0.5 or greater found in the VNIR data. Hence the data need further processing for meaningful representation.

### 3.1 Estimating Radiant and Kinetic Temperature

For estimation of temperature by remote sensing, the radiation intensity emitted from the target (heat source) is used. Planck's radiation equation can then be applied to convert measured spectral radiance to kinetic temperature. Planck's radiation equation for a blackbody is given by:

$$L_\lambda = \frac{2\pi c^2 h}{\lambda^5} \frac{1}{e^{\frac{hc}{kT\lambda}} - 1}$$

with $\lambda$ = wavelength, $L_\lambda$ = spectral radiance, $h$ = Planck's constant, $k$ = Boltzmann's constant, $T$ = Temperature [K], $c$ = speed of light

This can be rewritten as

$$T = \frac{C_2}{\lambda \ln[(C_1 \lambda^{-5} / \pi L_\lambda) + 1]},$$

where $C_1 = 2\pi hc^2 = 3.742 \times 10^{-16}$ Wm$^{-2}$, $C_2 = hc/k = 0.0144$ mK

Thus the temperature $T$ of a blackbody can be estimated from spectral radiance at a particular wavelength.

Once the corrected spectral radiance value ($L_\lambda$) for a pixel is known, it can be used for computing the temperature value. This temperature estimated from $L_\lambda$ is the radiant temperature $T_R$ as the effect of non-blackbodiness (spectral emissivity) is still to be taken under consideration. From the radiant temperature $T_R$ the kinetic temperature $T_K$ can be calculated. Commonly a spectral emissivity value of about 0.9-0.96 can be assumed for most terrain materials.

Utilizing the above concept and procedure, the temperature of various features and objects can be estimated from the thermal infrared data. The temperature obtained in this way represents the overall temperature of the pixel, and is called the pixel-integration temperature.

## 4. Thermal Infrared Satellite Imagery Analysis Techniques

### 4.1 Sharpening by Discrete Wavelet Transform

Wavelets provide an efficient means of representing high and low frequency components of multispectral images and can be used to perform image sharpening. The wavelet-based image fusion, as introduced by Ranchin and Wald [5], was originally developed for the pan-sharpening of the multispectral bands from high-resolution sensors, such as QUICKBIRD and IKONOS, and adapted here for the sharpening of TIR data (90m or 60m) to the resolution of the ASTER VNIR data (15m) or LANDSAT 7 panchromatic band (15m)

The discrete wavelet transform (DWT) of a two-dimensional image is equivalent to an iterative application of the high-low-pass filter bank illustrated in Figure 2. The application of the filter corresponding to the so-called Daubechies D4 wavelet to a satellite image $f_1(i, j)$ (1m resolution) works as follows. The high frequency information (wavelet coefficients) is displayed in the upper right, lower left and lower right quadrants, respectively. The original image with its resolution degraded by a factor two, $f_2(i, j)$, is in the upper left quadrant. Applying the filter bank iteratively to the upper left quadrant yields a further reduction by a factor of 2.

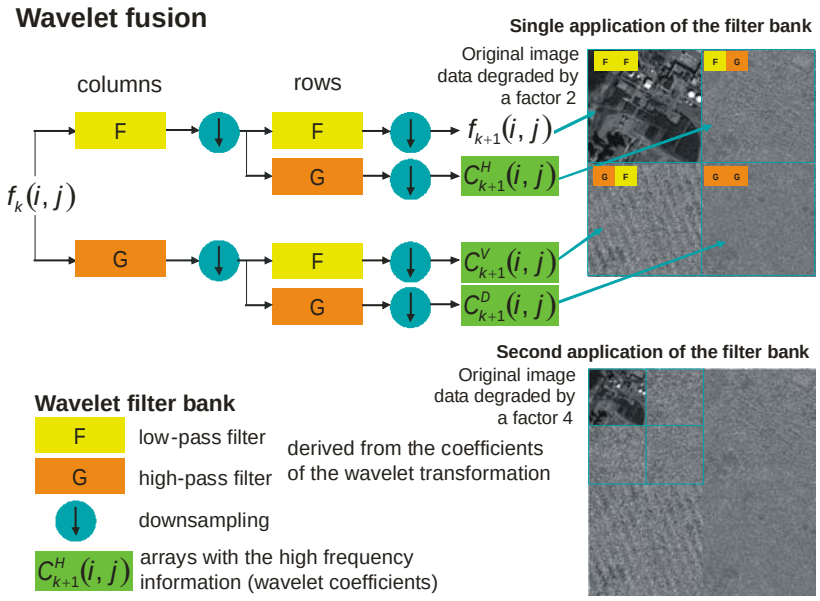



Figure 2: Wavelet filter bank. F is a low-pass and G a high-pass filter derived from the coefficients of the wavelet transformation. The symbol ↓ indicates downsampling by a factor of 2. The original image $g_k(i, j)$ can be reconstructed by inverting the filter.

The fusion procedure, in which the resolutions of high-resolution (panchromatic) and the low-resolution components differ exactly by a factor of 4, is then as follows: Both the degraded high-resolution image band and the low resolution image band(s) are compressed once again and the high frequency components are sampled to estimate the correction coefficients These coefficients, which are then used to normalize the wavelet coefficients for the high-resolution (panchromatic) image to those of the low-resolution image.

The degraded high-resolution panchromatic image $f_3(i, j)$ is then replaced by the each of the low resolution bands and the normalized wavelet coefficients are used to reconstruct the original high resolution. As a result, the fused bands will have thus obtained what would be seen if the multispectral sensors had the resolution of the panchromatic sensor.

The Discrete Wavelet Transform has been implemented as an ENVI extension by Canty [8].

### 4.2 Reed-Xiaoli (RX) Anomaly Detection

RX Anomaly Detection uses the Reed-Xiaoli Detector (RXD) algorithm to detect the spectral or colour differences between a region to be tested and its neighbouring pixels or the entire data set. This algorithm extracts targets that are spectrally distinct from the image background. For RXD to be effective, the anomalous targets must be sufficiently small, relative to the background. Results from RXD analysis are unambiguous and have proven very effective in detecting subtle spectral features. ENVI [9] implements the standard RXD algorithm:

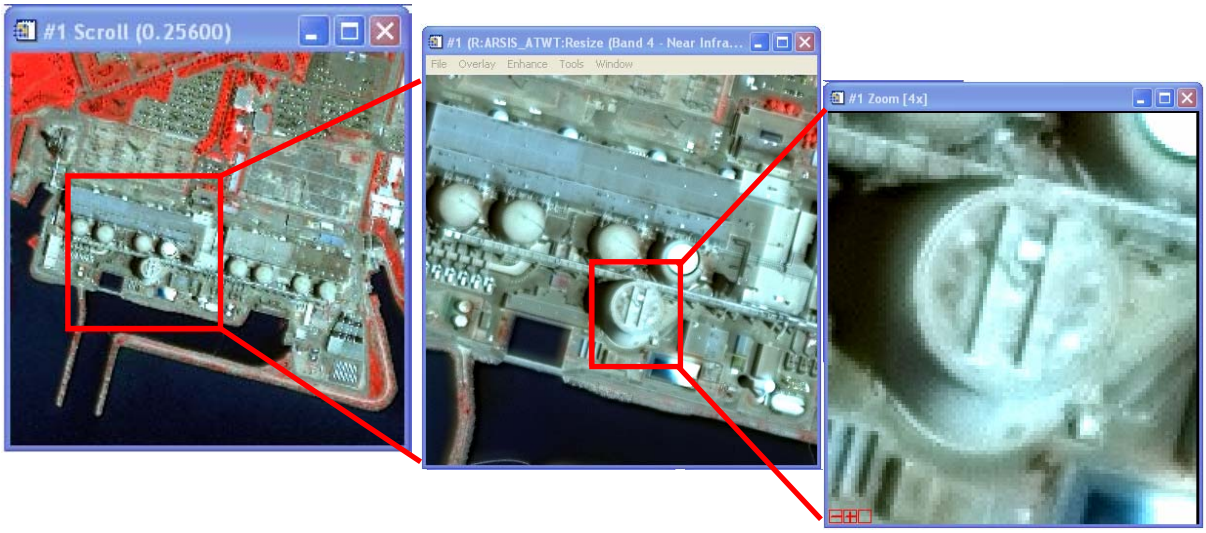
$$\delta_{RXD}(r) = (r-\mu)^T K_{LxL}^{-1}(r-\mu),$$

where $r$ is the sample vector, $m$ is the sample mean, and $K_{LxL}$ is the sample covariance matrix.

RXD works with multispectral and hyperspectral images. Bad pixels or lines appear as anomalous, but they do not affect the detection of other, valid anomalies. As with any spectral algorithm, exclusion of bad bands increases the accuracy of results. Currently, this algorithm does not differentiate detected anomalies from one another.

## 5. Investigations

The proposed procedures were applied to a LANDSAT-7 image acquired over Pickering, Canada in August 2002 and two ASTER imagery acquired over La Hague, France in July 2000 and December 2001. Figures 3 and 4 provide an overview for the two locations based on Ikonos data and GoogleEarth.




Credit: Space Imaging

Figure 3: Pan-sharpened Ikonos data over Pickering from July 20, 2000: NIR (R), Red (G), Green (B)


Figure 4: Overview of faculties at the La Hague site (Google Earth™)

For pre-processing, atmospheric correction was applied to the data sets and then the emissivity and temperature information in the radiance data measured were separated. As a result, the temperatures

can be displayed on a given scale (Figure 5). The surface temperatures given by the LANDSAT-7 image over Pickering was then fused with the 15m (Figure 6).

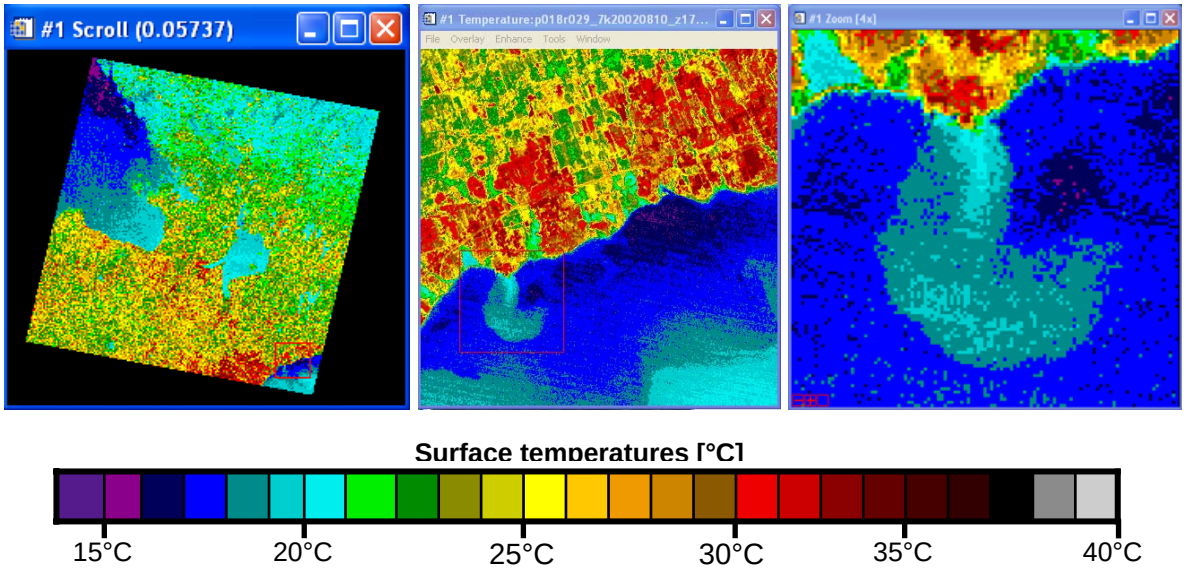



Figure 5: Surface temperatures given by LANDSAT 7, 10 August 2002; ~ 11:00am local time; 27°C air temperature; clear conditions

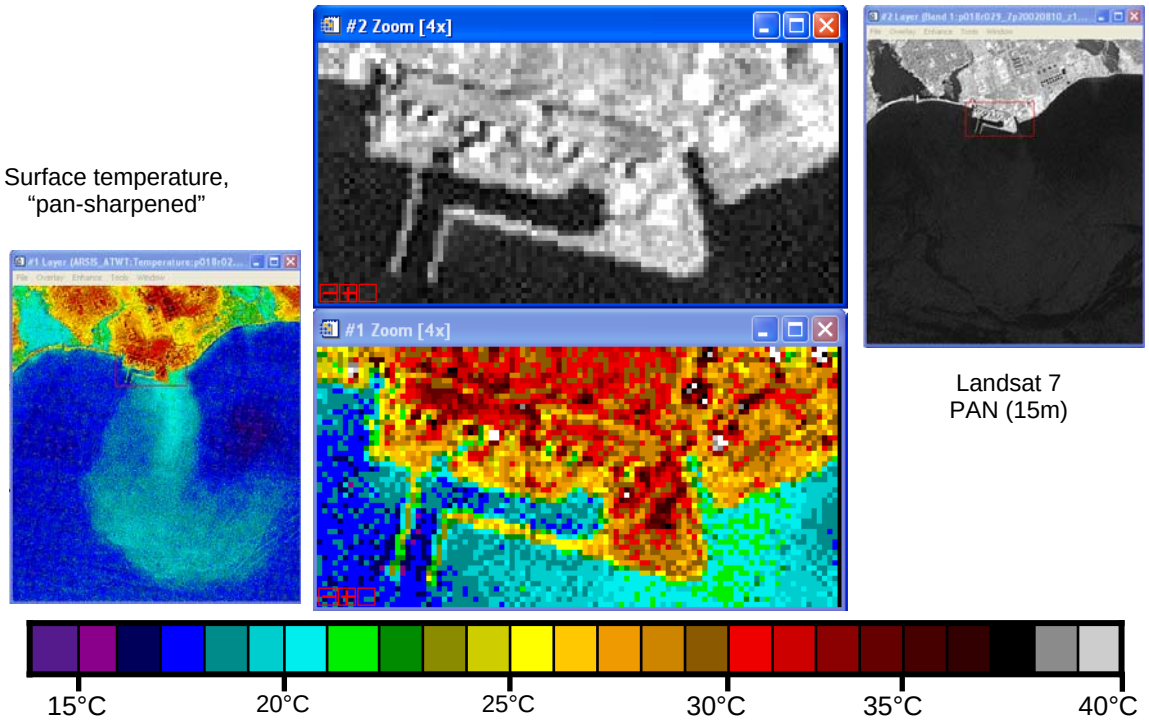



Figure 6: Surface temperatures fused with Landsat 7 PAN

The same procedure – atmospheric correction, conversion to emissivity and temperatures – was then applied to the two ASTER scenes over La Hague. For sharpening, one VNIR channel resampled to the resolution of 22.5m was used. Moreover, the thermal anomalies were calculated using the RXD technique. Figure 7 shows the result for July 2000, Figure 8 for December 2001. Due to the different thermal inertia of land and water, the thermal behaviour during winter and summer is converse. Whereas land quickly heats up and cools down, water absorbs and emits heat much slower. Thus, the

land surface temperature is much higher than the water surface temperature in summer (around noon) and colder in wintertime. The reprocessing plant indicates a higher surface temperature than the surroundings in both summer and winter, but only for the data acquired in July, the signal is very significant.

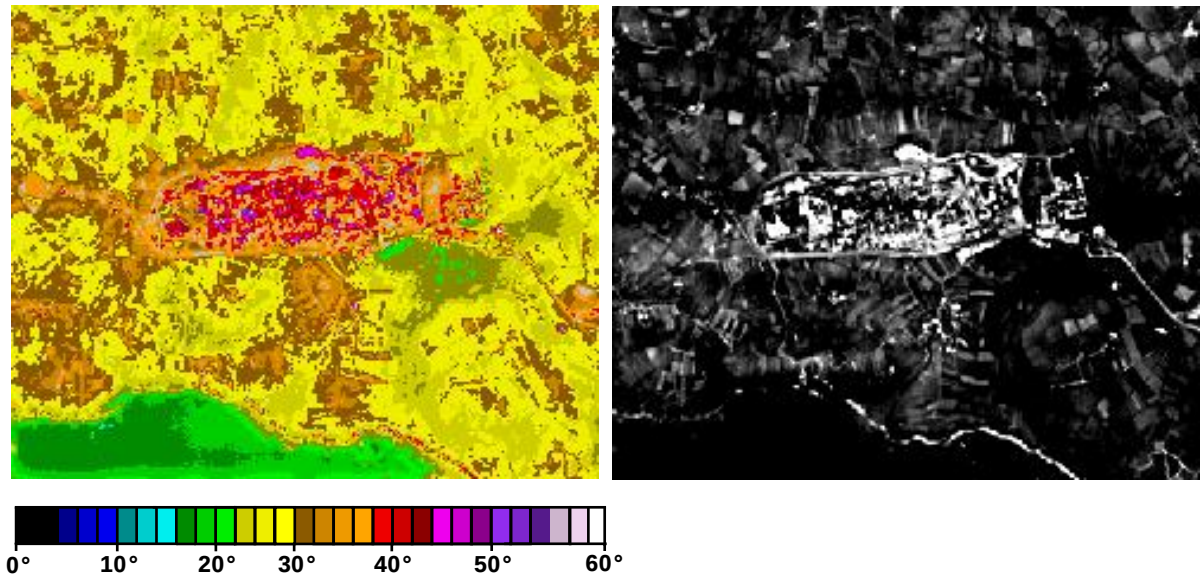



Figure 7: Temperatures given by ASTER [°C] in 19 July 2000, 11:37 am local time, colour-mapped (left) and positive temperature anomalies (right), both image data were fused to VNIR (22.5m)

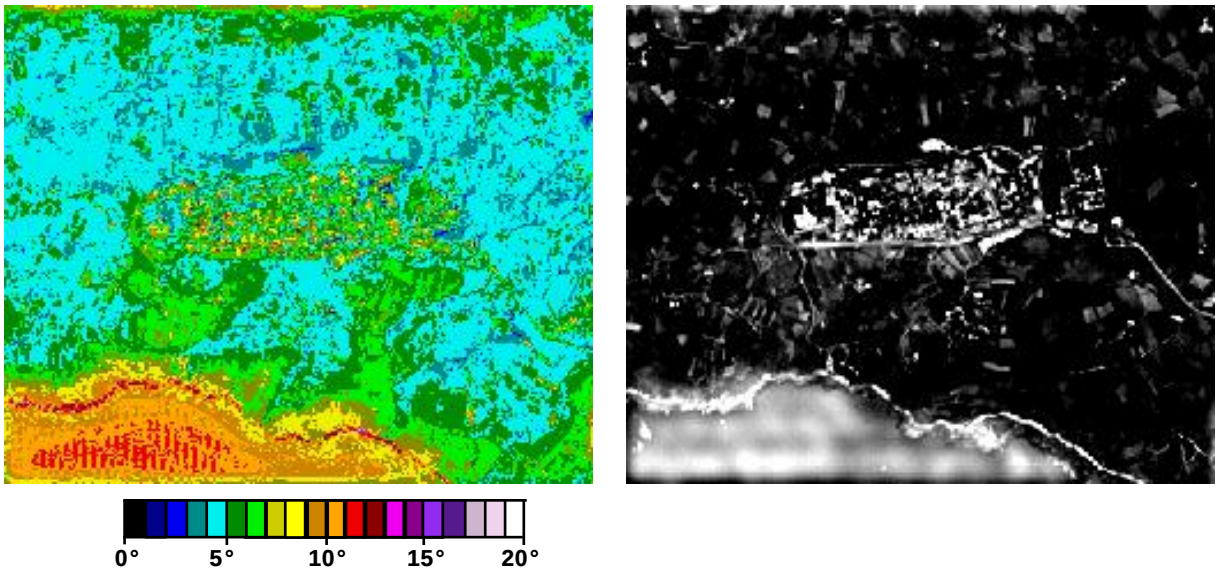



Figure 8: Temperatures given by ASTER [°C] in 15 December 2001, 11:13 am local time, colour-mapped (left) and positive temperature anomalies (right), both image data were fused to VNIR (22.5m)

## 6. Conclusions and outlook

Thermal infrared remote sensing data provides safeguards-relevant information, even though the spatial resolution is relatively low. After converting the thermal infrared data to emissivity and temperatures, image fusion (here: discrete wavelet transform) with bands of higher spatial resolution facilitates the interpretation of the temperatures. Using anomaly detection tool, such as the RX anomaly detection, are useful for extracting “hot spots” in a specific region or the whole scene.

Algorithms offering the detection of thermal anomalies in the sub-pixel domain, as proposed by Dozier [10] and Schäfer [11] are under investigation and show promises. Techniques comparing the temperature estimated through simulation (heating induced by topography, time of the day, land cover) with the temperature measured by the satellite sensor (heating due to other) are also currently being implemented.

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# Image Analysis based on object features and invariant moments exemplified for nuclear verification

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***Abstract:***

*When applied to high-resolution imagery, the traditional pixel-based image processing algorithms are often inadequate. Especially for monitoring or detecting small structural objects, object-based procedures are more effective and meaningful. The ability to include features such as the size of the objects, their shape or texture, their proximity to other objects and the relations between the objects observed different scales considerably extends the image analysis possibilities. On the other hand, image objects belonging to one object class may differ with respect to absolute dimension or orientation. For this reason it is necessary to determine invariant object features for classification. Another important source of information is the a-priori expertise knowledge which can be included within the recognition process.*

*This paper investigates the object-based image analysis approach for nuclear verification using high-resolution image data. The extraction of the objects from automatically pre-processed images takes place by segmentation, at which stage the primary segments should ideally represent real objects. Within the eCognition object-oriented classification environment, the initial selection of object features is carried out by an analysis tool, SEparability and THreshold (SEaTH). This provides the basis for image classification within eCognition. Classification (including the relevant object features provided by eCognition and expert knowledge) is then carried out by means of a ruled-based classification model. The accuracy and robustness of the methodology are investigated with test cases involving various high-resolution satellite images.*

**Keywords:** nuclear verification, safeguards, object-based image analysis, high resolution image data, automation

## 1. Introduction

Since the availability of high-resolution satellite imagery, the use of remote sensing data has become very important for nuclear verification and safeguards purposes. Due to technical improvements regarding the spatial and spectral resolution, satellite imagery can now form the basis of complex systems for recognizing and monitoring even small-scale and short-term structural features of interest within nuclear facilities.
Within the framework of nuclear verification, satellite data can by used to support various safeguards objectives. The data can aid the process of proofing site declarations of member states, disclosure of clandestime activities (new sites or within existing sites), pre-selection of targets for visual inspection or the detection and classification of changes.

When adapted to high-resolution imagery, the traditional pixelbased image processing algorithms are sometimes limited. Especially if small structural objects are to be detected, object-based procedures seem to be more precise and meaningful. In comparison to the purely spectral-based features used within the pixel-based approaches, the inclusion of features such as the size or orientation of an object, its shape or texture and its relations to other objects on the same or at different scales, considerably extends the possibilities for image analysis. This paper proposes a new combined object-based image analysis and change detection methodology for nuclear verification.

## 2. Object based image analysis

Computer-based, object-based image analysis is, in a first approximation, comparable to visual perception. An image interpreter recognizes, along with the color of an image, also shapes, textures and coherent regions present within it, and associates meaningful objects with them. A similar goal is pursued in object-based image analysis, although the complexity and effectiveness of human perception are, of course, far from being achieved. In the following we describe briefly the used workflow, see Fig.1 for a schematic overview. For a comprehensive description of the single steps, see the quoted literature.

### 2.1. Workflow

The arial images used in this paper require no pre-processing. On the other hand Quickbird images where pre-processed in an automated three step procedure. The first step is image data fusion via a panchromatic sharpening technique. Here a wavelet based algorithm is applied, which produces good spatial resolution without significantly affecting the original spectral values. This fusion is necessary, because the original Quickbird image data contain a lower-resolution multi-spectral data set (2.4 m) and a high-resolution panchromatic image (0.6m). For the object based image analysis a high-resolution multi-spectral data set with 0.6m ground resolution is a great advantage [1].

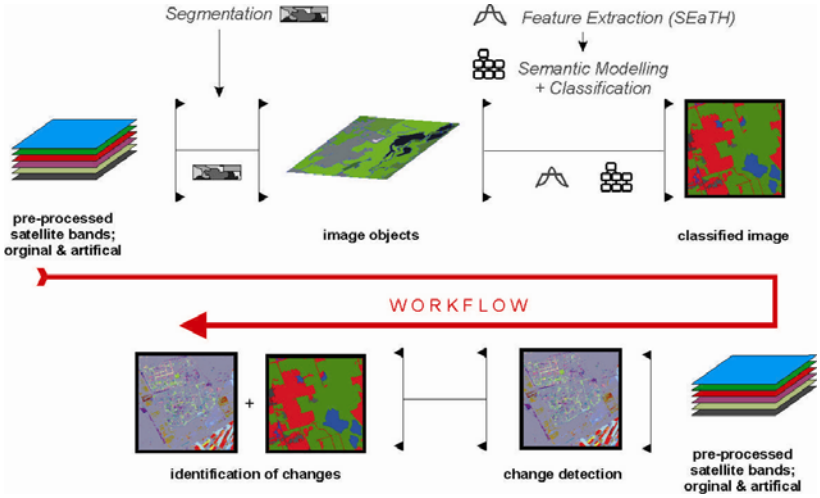


***Figure 1: General workflow of the object based image analysis***

The second pre-processing step is an automated image-to-image registration. A precise geometric correction is essential for an exact pixel-by-pixel comparison during the change detection process. Here an image-to-image registration is carried out based on image correlation algorithms with sub-pixel accuracy (RMS error well below +/- 1 pixel) [2].

Radiometric correction procedures (step three) are necessary to obtain absolute surface radiance or reflectance by removing atmospheric effects. For transferability of image classification models or change detection applications of satellite imagery, absolute atmospheric modelling is seldom needed. Assuming that the relationship between the at-sensor radiances measured at two different times can be approximated by linear functions, a relative radiometric normalization seems to be sufficient. Here, a relative radiometric normalization based on the time-invariant pixels is applied to the image data [3].

The extraction of the objects from the pre-processed images takes place at the lowest level by segmentation, at which stage the primary segments should ideally represent the real world objects. Here the multiresolution segmentation algorithm of the software Definiens Professional [4] is used. It has been shown that the segmentation algorithm used is one of the currently most efficient algorithms for remote sensing [5]. Feature recognition is carried out by the analysis tool *SEparability and Threshold (SEaTH)* [6] [7] providing, together with expertise knowledge, the basis for image classification. After developing a ruled based classification model within the eCognition environment, the classification is carried out.

For change detection purposes a new combined change detection methodology is applied. This technique uses an existing pixel based change detection algorithm, the Multivariate Alteration Detection (MAD) [8]. Tthis algorithm shows where significant changes occur over time in different multitemporal images. Combining this information and the previously achieved classification result, an automated detection and interpretation of the changes is possible. This means that the methodology is able to figure out where changes occur, what has changed and to differentiate between different kinds of changes [9].

## 3. Automated image classification and change detection

### 3.1.1. Esfahan, Iran

The Nuclear Fuel Research and Production Center (NFRPC) is Iran's biggest nuclear research center, built in 1974, south-east of the city of Esfahan. At NFRPC, there is a miniature neutron source reactor (MNSR), a light water sub-critical reactor (LWSCR), a heavy water zero power reactor (HWZPR), a graphite-moderated sub-critical reactor (GSCR), a fuel fabrication laboratory (FFL), a uranium chemistry laboratory (UCL), a uranium conversion facility (UCF) as well as a fuel manufacturing plant (FMP). Some of these facilities are still under construction, in operation or shut down. Iran is a party to the Non-Proliferation Treaty (NPT) and, at present, IAEA Safeguards measures are being applied at NFRPC.
Fig. 2 shows the two pre-processed Quickbird data sets for the acquisition periods July 2003 (left) and May 2004 (right).


**Figure 2: Images of the pre-processed NFRPC Esfahan from July 2003 (left) and May 2004 (right)**

Site monitoring was carried out with ASTER satellite data at 15m ground resolution, and, for observation of individual facilities over time, high spatial resolution Quickbird images at 0.6m ground resolution were used. The case study used Quickbird images acquired over NFRPC on an annual basis, in this case, July 2003 and May 2004. The whole monitoring procedure, including pre-processing, modelling, image classification and change detection, cannot be presented here in detail.

Rather, the focus will be on the results of automated pre-processing, object-based image classification, and automated change detection from July 2003 to May 2004.

The workflow of the image analysis including combined change detection is described in the previous section. Extraction of objects was carried out by means of multi-scale segmentation. This led to a hierarchy of image objects, the individual features of the Esfahan site being identified at the lowest segmentation level. By increasing the scale parameter the objects become coarser, until the structures of interest are included in one object. For segmentation, the multi-resolution segmentation algorithm of the commercially available image analysis software eCognition was used with standardized parameters. Feature extraction and semantic modelling were implemented on the basis of a feature analysis with the SEaTH method for each of the following defined object classes: Buildings, Background, Roads, Shadows, Vegetation and Walls-Fences-Pipe Systems. This led to a ruled-based, object based image classification model for NFRPC. The model itself will not be presented here, however, Fig. 3 shows the result of the image classification for the NFRPC 2003 (left) and 2004 (right).


**Figure 3: Classified images of the NFRPC Esfahan from July 2003 (left) and May 2004 (right)**

At first glance, the images look spatially coherent, with none of the salt & pepper effects that often appear in pixel-based approaches. Most of the objects are classified correctly. The overall classification accuracy for the 2003 image is 89%, and for the 2004 image it is 87 % (see [10] for more information about the assessment of the accuracy). As the study focuses on the detection of undeclared buildings and other man-made structures, the so-called Producer's Accuracy was determined for the class Building Development. The Producer's Accuracy measures how well a certain class has been identified. For both images the result was between 92% - 98%.
The combination of pixel-based change detection and object based image classification has been proven to be a viable method to detect and identify significant changes in multi-temporal data. Figure 4 shows a result of this change detection methodology.

The Figure builds up the significant changes in the time period July 2003 to May 2004 for NFRPC. The changes where identified for the single object classes. Most of the changes have occurred within the Buildings and Vegetation classes. Focussing on the man made objects, a subset of the facility is extracted and shown in the white frame. This subset differentiates between several kinds of change, represented by different MAD components (see [8] for an explanation of the MAD components). The most dominating change classes are the MAD 4- and the combined class MAD3+4-. The class MAD 4- builts up the changes resulting from sun reflectance of the roof. The class MAD3+4- show the new constructed buildings. The red frame shows a comparison of the two acquisition times.


**Figure 4: Comprehensive change detection result for the NFRPC Esfahan. The background image shows all significant changes in the time period 2003 to 2004. Within the white frame different kind of changes are discriminated. The red frame shows the detect new constrcted buildings and a comparison between 2003 and 2004.**

## 4. Implementing of expertise in the image analysis

This section deals with the computer-based interpretation of high-resolution arial images of Nuclear Power Reactors. The whole work flow, containing segmentation, feature analysis, modelling and classification is described in Sect. 2. In addition to the previous methodology also expert knowledge is included in the classification model. The expertise is based on the so called ‘key- features’ developed by Jasani [11]. These key features are mostly shape features describing special buildings of nuclear facilities. These buildings could be e.g. the reactor dome, the generator hall or cooling towers. The next section checks the applicability of this visually achieved ‘key features’ to a computer based image analysis. If the methodology works, special buildings of the nuclear fuel cycle should be detectable.

### 4.1.1 Grohnde NPP

In this section a German nuclear power plant (NPP) is investigated, the Grohnde NPP. The Grohnde NPP is a Pressurized Water Reactor type with 1,366 MW capacity. It is located on the Weser bank

between Kirchohsen and Grohnde. The commercial operation started in 1985. It has two natural draft cooling towers which use the Weser water. Fig. 5 shows a orthophoto of the facility taken on 28.05.2005. The image is provided by the 'Landesvermessung + Geobasisinformation Niedersachsen (LGN)' and has a ground resolution of 0.40 cm.


**Figure 5: Orthophoto of the Grohnde NPP, may 2005.**

Again the segmentation algorithm of the software eCognition is used to generate the image objects. As an additional information layer the AK5 cadastral map is included in the segmentation process. The information of the cadastral map is weighted by a factor of 10. The algorithm extracted about 1700 image objects. To have only one scale level for the classification the real world cooling tower objects where manually fused from three to one complete image object.

First of all a classification is carried out, based only on the features derived with SEaTH. The information from the AK 5 is not included in the classification process. The working procedure is the same as before. The defined object classes are: Buildings, Fields, Meadows, Roads, Shadows and Water. The classes Buildings and Roads are defined as child classes of the parent class Sealed Surface. First the class Sealed Surface is determined and then this is split up in the classes Buildings

and Roads. The features used in the parent class are automatically passed down to the both child classes. This first classification result is not shown here.

The next step is the application of the "interpretation key" derived from Jasani [11]. With the key one trys to identify special kinds of objects within the Buildings class, such as the Reactor Dome, the Turbine \& Generator Hall, the Reactor Control and the Natural Draft Cooling Towers (ND Cooling Towers). All of these objects belong to the Nuclear Fuel Cycle (NFC).

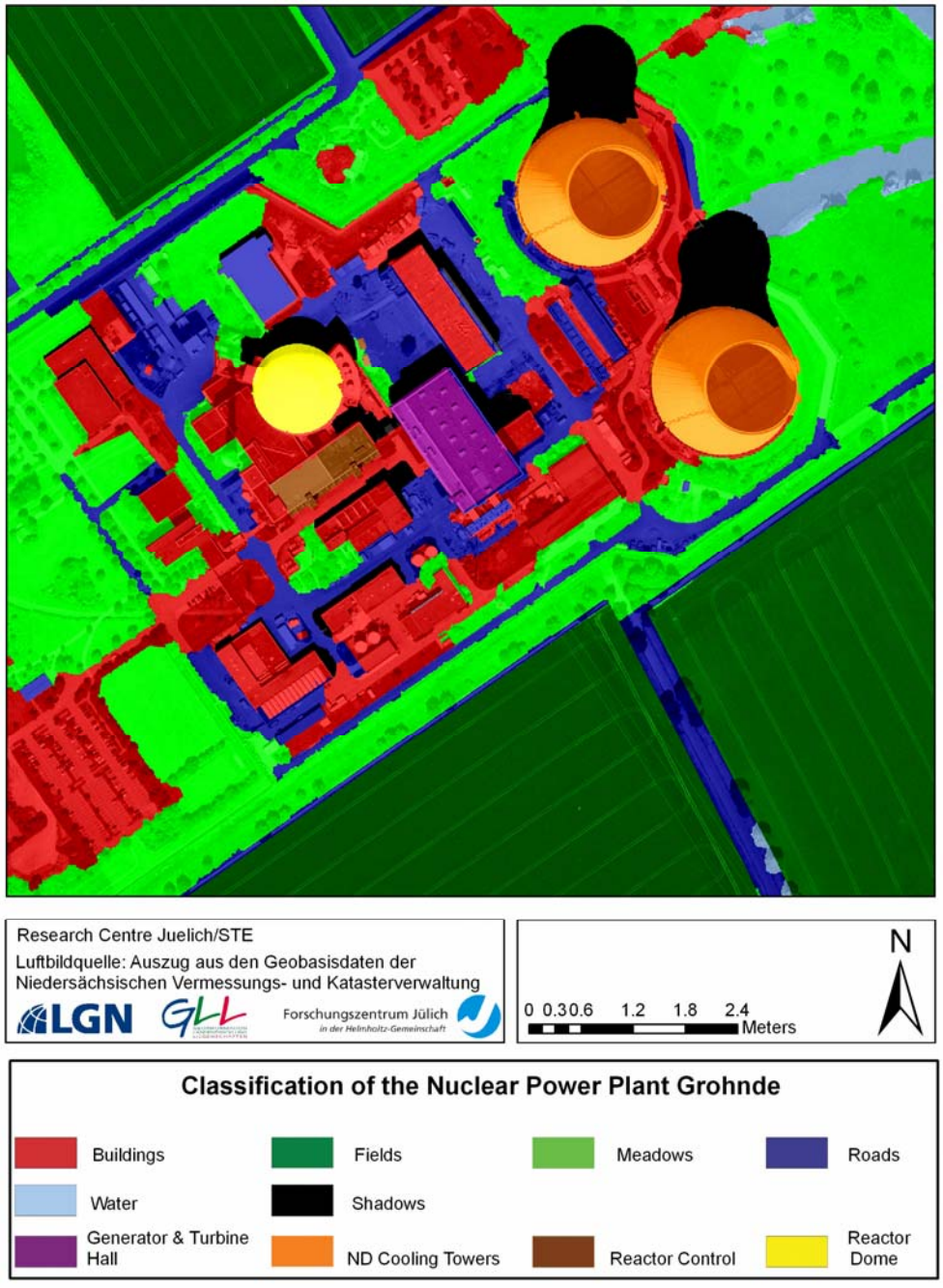


**Figure 6: Classification of the Grohnde NPP including expertise features**

Characteristic features for the class Reactor Dome are its round shape and a diameter of about 60m. The objects corresponding to the class Turbine & Generator Hall are near to the reactor dome and have a dimension of approximately 50m * 90m. The ND Cooling Towers are near to water, have a round shape and a diameter of about 120m. For the Reactor control a rectangular building very close to the reactor is named in the key. These features were implemented in the classification model. These new classes are modelled as child classes of the class Buildings.

As one can clearly see in Fig.6 the use of additional key features leads to an information benefit. Now special buildings within the Buildings class could be identified. The classification model detected in addition to the already known classes Buildings, Fields, Meadows, Roads, Water and Shadows also image objects of the classes Generator & Turbine Hall, ND Cooling Towers, Reactor Control and Reactor Dome.

## 5. Conclusion

For nuclear safeguards purposes a methodology was developed to facilitate identification of objects and detection of significant changes at complex nuclear facilities by evaluating high-resolution satellite imagery. The presented techniques allow a automated pre-processing of high-resolution data. Identification of objects at nuclear facilities was achieved by object-based image analysis, including standardized segmentation, statistical feature analysis and rule-based image classification. The results of image classification and change detection were very satisfying for all the case studies described here.

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# Identification of key features of nuclear facilities for interpretation of imageries from remote sensing satellites – *Enrichment and reprocessing plants*[*]

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## Abstract

The uranium enrichment process has been the only part of the nuclear fuel cycle which has been kept under strict secrecy and control. The reasons are obvious. All the five nuclear weapons states have used U-235 in their first fusion devices and three weapons states, including the two new ones, have used U-235 in their first fission bombs. While highly enriched U-235 has been used to start the fusion reaction in the thermonuclear weapons, Pu-239 would also trigger such a reaction. Should a country decide to start its nuclear weapons (fission weapon) programme with U-235 as the fissile material, then the acquisition of an enrichment facility becomes essential. The uranium for a nuclear weapon is enriched to at least 50 per cent and it is generally assumed that the fissile material used by nuclear weapon powers in their uranium weapons contains uranium enriched to at least 90 per cent.

Uranium enrichment is an essential part of the light water reactors (LWRs), advanced gas-cooled reactors (AGRs) and high-temperature reactors (HTRs). Of these, the LWRs are most widely used. Thus, even if a country does not embark upon a nuclear weapons programme, the development and the possession of an enrichment facility would make the country independent of other sources of enriched uranium fuel for its nuclear power programme, particularly if the country has uranium resources of its own.

The natural or enriched uranium fuel is made into reactor fuel rods and placed into the reactor. The fissile material that may be produced in the reactor is safe from theft or governmental diversion as long as the fuel elements are in the reactor. Outside the reactor, the fissile material can be diverted if the spent fuel is reprocessed. This, the second sensitive part of the nuclear fuel cycle, is considered in this paper.

It is, therefore, important to examine the characteristics of enrichment and reprocessing facilities so that a key could be developed to aid interpretation of images from remote sensing satellites.

## 1. Introduction

The uranium enrichment process has been the only part of the nuclear fuel cycle that has been kept under strict secrecy and control. The five nuclear weapons states have used uranium-235 (U-235) in their first fusion devices and three weapons states, including the two new ones, have used U-235 in their first fission bombs. Acquisition of an enrichment facility becomes essential if a country decides to start a nuclear weapons (fission weapon) programme with U-235 as the fissile material. Uranium for a nuclear weapon is enriched to at least 50 per cent and it is generally assumed that

the fissile material used by nuclear weapon powers in their uranium weapons contains uranium enriched to at least 90 per cent. As most power reactors use enriched uranium, the development and construction of enrichment facility becomes attractive from commercial point of view also.

Several methods for isotope separation were known even before nuclear fission was discovered. The aim was to increase the proportion of fissile U-235 atoms within uranium. The most common ones, considered in this paper, are the centrifuge and gaseous diffusion processes.

A number of large commercial enrichment plants are in operation in China, France, Germany, the Netherlands, Russia, the UK, and the USA. Pakistan and Brazil have smaller ones in operation. It is, therefore, important to examine the characteristics of enrichment facilities so that a key could be developed to aid in the interpretation of images from remote sensing satellites by an interpreter as well as automatically by a computer, particularly in an image that covers a large area around a facility.

The other fissile material that can be used in a fission weapon is plutonium-239 (Pu-239). This does not exist in nature like U-235 but it is produced in a nuclear reactor. If a plutonium root is chosen for ones nuclear weapons programme then it is important to have a reactor fuel reprocessing capability or an access to a reprocessing plant. Thus, it is also important to determine a "key" for such a plant. Therefore, in the second half of the paper, a key is developed for such a facility.

Consider first a gas centrifuge enrichment facility.

## 2. Centrifuge enrichment plant

In a gas centrifuge process, uranium hexafluoride ($UF_6$) gas is spun in a cylindrical chamber at a very high speed. This causes the slightly heavier isotope U-238 to separate from the lighter U-235 and is drawn towards the bottom of the chamber and extracted whereas the lighter U-235 collects near the centre from where it is also removed. The enriched U-235 is then fed into another centrifuge. The process is repeated many times through a series of centrifuges known as a cascade.

After an examination of a number of aerial images of uranium centrifuge facilities, it was found that such a complex generally consists of two major linked buildings. One is the uranium separation facility consisting of centrifuge cascades, such systems as $UF_6$ feed and depleted uranium removal devices, the ventilation and the service systems to operate the facility and the central control room from which the whole facility is monitored and operated.

The second building contains the technical infrastructure that includes an area in which $UF_6$ of different enrichment assays can be blended and the processed uranium storage room. This building could also house laboratories, workshops, facilities to clean $UF_6$ containers, pumps, valves and equipment to treat the resulting liquid effluent. Then there are a number of other buildings that depend on the design of the facility. For example, there may be an emergency fossil-fuel power plant; $UF_6$ feed delivery and storage facility; and a switchgear station. Often the processed and depleted $UF_6$ containers with fuel are stored outside in the open.

The $UF_6$, which is in solid form at room temperature, is brought to the enrichment building in cylindrical steel containers about 3.8m long and 1.2m diameters by rail. The containers are placed in heating chambers and heated to about 70°C by electrically heated air. The feed pipes are connected to the centrifuges to channel the vaporised $UF_6$ into the cascades. The ventilation system is such that fresh air can enter the hall but air from it passes through the filters in the ventilation systems and into the atmosphere. In the main cascade halls are situated a number of centrifuges that are connected in parallel. A separate de-sublimer is used for each group of cascades so that the uranium of different U-235 concentrations, from each cascade hall is collected separately. Alongside the cascade are cabinets housing the controls for the cascades and the electricity supply required to drive the centrifuges.

After leaving the cascades, the enriched and the depleted $UF_6$ are collected separately in de-sublimers. These are steel vessels that can be cooled to -70ºC causing the $UF_6$ to solidify. This temperature is achieved in the refrigeration system. The full de-sublimer is then heated to about 50°C. This is achieved by hot water and heat exchangers and the heat transfer fluid needed for the de-sublimers. The enriched product and the tails are piped from the de-sublimers into separate transport containers in cooling chambers where the enriched products are cooled by air and the tails by cooled water. The water is cooled to 6ºC by air-cooled chillers. The final heat rejection is achieved via high capacity Freon chillers discharging heat to the atmosphere via roof mounted air-cooled radiators. A typical 1,000 tSWU/a plant would dissipate about 15 MW of heat via roof-mounted air cooled radiators or a low-profile forced draught cooling tower.

As the required energy for a uranium centrifuge plant is considerably less than, for example, that needed for a diffusion plant, electricity can be supplied from the public grid. Thus, no large power plants are associated with centrifuge enrichment facilities. However, an important component of a centrifuge device is power supply frequency converter. Power supply from the grid has a frequency of 50- or 60Hz while a much higher frequency, typically over 600Hz, is required to operate the high speed centrifuges[1]. Thus, apart from the two main buildings, a centrifuge plant complex also has transformers and a switchyard.

The floor size of a centrifuge plant may depend on the number of centrifuges deployed. It would not be surprising to find a linear relationship between the floor area and the capacity of such a plant. This was shown to be the case.[2]

A centrifuge enrichment facility is usually enclosed within perimeter fences. There are possible railway tracks or extensive road network required to bring in $UF_6$ drums to the feed plant.

Thus, following “key” for a centrifuge enrichment facility could be constructed:

- A main gas centrifuge cascade hall is usually connected to a second building that houses the technical infrastructure facility;
- The size of the cascade hall will depend on the number of cascades, as it is proportional to the number of centrifuges;
- The excess energy is discharged into the atmosphere via roof-mounted air-cooled

radiators or a low-profile forced draught cooling tower;

- The second building houses the fuel feed and delivery facility;
- While the gas centrifuges do not consume much electricity, there is still a need for some energy that is obtained from the national grid so that there is an area containing the required transformers and switchyard;
- There is an outside storage area for depleted $UF_6$ containers;
- A railway line or a road associated with the plant for transporting $UF_6$ for enrichment and for the depleted $UF_6$;
- The whole complex is within a perimeter fence.

Both aerial and satellite-based images were analysed and in the following sections only the images acquired by satellites over a uranium centrifuge facility are described to indicate the validity of the above key.

### *Japanese Rokkasho uranium centrifuge enrichment plant*

The Rokkasho uranium centrifuge enrichment plant, located in the north-eastern part of Japan, is owned by the Japan Nuclear Fuel Ltd., and has been in operation since 1992. Initially the production capacity was 600 tons SWU/a. It was expanded to 1,050 ton SWU in 1999. In Figure 1, a detailed map of the Rokkasho facility is shown.

A Quickbird satellite image over the Rokkasho plant was acquired from the Google Earth site over the Internet: http://maps.google.com/maps. The image is shown in Figure 2. Two enrichment cascade halls, the technical infrastructure building, the transformer and the switchyard area and the feed and the depleted $UF_6$ storage area are identified and shown as the enlarged sections. The roof mounted cooling system can be seen clearly in the satellite image over the two cascade halls. From the scale provided in the Google Maps image, the sizes of these halls were determined and found to be approximately 11,500$m^2$ (the building on the left) and 19,400$m^2$ (the building on the right).

### *Brazilian Resende uranium gas centrifuge plant*

Consider the gas centrifuge facility in Brazil located some 20km west of Resende in the Rio de Janeiro State. The site is near the east cost of Brazil. Figure 3 is a satellite image with a map superimposed over it showing the location of the enrichment plant.

The Resende Enrichment plant, operated by Indústrias Nucleares do Brasil (INB), was inaugurated on 5 May 2006. The completion of the plant is scheduled for 2010. Once in full operation, the plant could provide 60% of the fuel for the Angra reactors. The plant uses national technology developed by the Brazilian Navy. The commercial start-up of the Resende enrichment plant, scheduled for 17 January 2006, was delayed. At present, the plant is running in test mode since August 2005. The completion of the first stage of the plant (114,000 SWU/year) has also been postponed from 2008 to 2010.[3]

A Quickbird satellite image over the Resende plant was acquired from the Google Earth site over the Internet: http://maps.google.com/maps (see Figure 4). It can be seen that the above features under the “key” (perimeter fence, cascade hall, cooling towers, electrical switchyard and depleted $UF_6$ storage area) can be identified in the image. This facility is an example of

cooling towers being used rather than roof mounted air cooling system.

## 3. Diffusion enrichment plant

In a gas diffusion method, the gaseous $UF_6$ is passed through a porous barrier material. The lighter molecules containing U-235 penetrate the barrier slightly faster. For $UF_6$, the difference in velocities between molecules containing U-235 and U-238 is small (~0.4 percent)[4] so that the amount of separation achieved by a single stage is very small. Therefore, a large number of porous barriers arranged in cascade are required to achieve any significant degree of uranium enrichment.

A typical gaseous diffusion enrichment plant may consist of a number of buildings. One of these is where the solid uranium hexafluoride ($UF_6$) in cylindrical containers is brought from the uranium conversion plant. In this, the feed plant, the $UF_6$ is heated to form a gas. The container becomes pressurized as more and more $UF_6$ gas is generated.

A second facility, the largest in the gaseous diffusion enrichment complex close to the above, is the building containing the gas diffusion assembly. Pipelines link both the buildings. The $UF_6$ gas is pumped through to the diffusion chambers containing porous membranes. Under pressure, the $UF_6$ gas diffuses through a series of several hundreds of such porous membranes. Each stage consists of a compressor, a membrane and a heat exchanger to remove the heat of compression. The excess heat, that can be considerable, is discharged to the environment via a number of cooling towers. The enriched $UF_6$ is withdrawn from one end of the cascade and the depleted $UF_6$ is removed from the other. Nearly 2,000 stages are required to obtain a concentration of 3% to 4% U-235. Hence the enrichment buildings of such plants are very large.

A gaseous diffusion process consumes considerable amount of energy that is provided by either conventional fossil fuel or a nuclear power plant. It can consume about 2,500 kWh (9,000 MJ) per SWU, while modern gas centrifuge plants require only about 50 kWh (180 MJ0 per SWU.[5] This requires a very large switchyard containing a number of transformers depending on how many diffusion stages are deployed. The switchyard may also contain three-phase (345 Kilovolt) gas circuit breakers.[6]

At the end of the process, the enriched and the depleted $UF_6$ gases are removed through a number of pipes to a third building known as the product withdrawal facility. The $UF_6$ gas is condensed back into a liquid that is poured into containers. It is then allowed to cool and solidify before transporting it to fuel fabrication facilities.

Considerable amount waste is generated from either cleaning various equipments used in the process or when they are replaced. Moreover, spent solvents generated at the site, and other contaminants need also to be stored. Most of these are disposed of by burying them on site and some stored in storage buildings.

This process has been used in the USA, Russia, the UK, France, China and Argentina as well. Today only the USA and France use the process on any significant scale. At present the gaseous diffusion process accounts for about 40% of world enrichment capacity.[7]

From the above it is apparent that, for a gaseous diffusion enrichment plant, there are a number of key features that could be identified:

- The enrichment complex is situated in a large area bounded by perimeter fences, particularly that used for defence purposes;
- Among several buildings, a facility, the feed plant, is where solid $UF_6$ from uranium conversion plant is brought to change the solid $UF_6$ into gaseous form;
- A very large building that houses, among other things, several hundreds of diffusion membranes, compression pumps and cooling systems;
- A number of cooling towers close to the enrichment facility;
- Because of the large power required, there is either a conventional fossil fuel or a nuclear power plant close to the enrichment facility;
- Close to the actual enrichment plant is a switchyard with a large number transformers;
- Power lines connecting the power plant and the enrichment facility;
- A perimeter fence;
- A third building, the product withdrawal facility connected with the enrichment building by a number of pipelines;
- As considerable amount of waste is produced in the enrichment process, it is either stored in large tanks, if it is in liquid form, or it is buried or stored in storage buildings on the site; and
- There is an open feed and depleted $UF_6$ storage area.

A number of gaseous diffusion plants were investigated using relatively high spatial resolution images in order to see if some of the above features could be identified in the images. Below the results of two sites, the US Paducah and the French Pierllate enrichment facilities are described.

***The US diffusion facilities***

Two Google Earth sites over the Internet: http://www.google.co.uk/ and http://maps.google.com/maps were used to acquire relatively high spatial resolution images to show the extent of details that can be seen from such images. It should be pointed out that while it was possible to download the images available on the Google Earth's website, the dates and several image characteristics are not available at this source. However, an advantage is that such images are cost free and they have been useful for the purpose of this study.

At present Paducah Plant is the only gaseous diffusion plant in operation in the United States. In the 1960s, the role of the Paducah plant changed from enriching uranium for nuclear weapons to one focused on producing fuel for commercial nuclear power plants. The Paducah Gaseous Diffusion Plant is located in south central Ohio, approximately 32 kilometres southwest of Portsmouth, Ohio, and 112 kilometres south of Columbus, Ohio. The site occupies just over $15km^2$ of land.

In Figure 5 an overview of the Paducah facility acquired probably by the US Landsat satellite is shown. The enrichment facility and the associated large conventional power plant are identified. Figure 10 shows the close up of the conventional power plant near the US Paducah enrichment plant in a QuickBird image. The barges carrying coal, the coal storage site, turbine and steam generators and various other associated buildings can be identified. The pylons carrying

electrical power lines could be traced to the Paducah enrichment facility south of this power plant. Clearly this plant provides the large electrical power required for the enrichment facility.

The Figure 6 is an enlarged section from Figure 5 showing the Paducah enrichment complex. Various features such as the possible storage area for the fuel caskets **(A)**, the switchyards **(B)**, the four enrichment halls **(C)**, the perimeter fence **(D),** the cooling towers **(E)**, the possible feed plant **(F)** and the possible waste burial area **(G)** can be identified.

Thus, it can be seen that the "key" described above can be used to interpret an image acquired from a satellite. This is possible even at a modest spatial resolution.

From the Figure 7, it can be seen that all the key features described above for a gas diffusion enrichment plant can be identified. In a larger image of the area the perimeter fence has also been identified. The "key" also holds for other plants, for example, the French Pierrelatte diffusion plant.

Thus, it can be seen from above analyses of the images that the "keys" suggested for both the gas centrifuge and the diffusion facilities can be used in the interpretation of satellite based remote sensing images.

## 4. Reprocessing

All power reactors in operation today produce plutonium-239 (Pu-239) as a result of nuclear reactions in the reactor fuel. In order to separate the unused uranium and plutonium from the fuel, reprocessing is required. This is a chemical operation which separates useful fuel for recycling from nuclear waste. Initially the spent fuel is brought from a rector to be stored in large water pools until they have cooled down for safe handling. The cooling pools are located in one section of the reprocessing plant. Then the used fuel rods have their metallic outer casing stripped away in a separate section of a reprocessing building. These are then dissolved in hot nitric acid in another part of the reprocessing plant. This produces uranium (96%), which is reused in reactors, highly radioactive waste (3%) that is stored either onsite or, once safe to handle, elsewhere, and plutonium (1%). If the fuel with short burn up times is processed, then weapons grade Pu-239 can be produced with minimum amounts of the other isotopes of plutonium being present.

Thus, fuel reprocessing is a part of both the civil or military fuel cycle. While a number of cases were studied, but in the following sections only two examples are discussed.

### *UK's reprocessing plant at Sallfield*

Various stages of reprocessing facility can be identified in an aerial photograph and in satellite imageries with moderate ground resolution. As an example, consider the UK reprocessing facility at Sellafield. Figure 8 is an aerial photograph of this plant.

An extract from a larger image over the Sellafield facility acquired by the French SPOT satellite on 19 March 1990 is shown in Figure 9. This image indicated that the spatial resolution may be adequate to use computer based automatic facility detection to be used.[8] Once the site of interest is detected, an image with high spatial resolution could be used to describe the site in some details. This is indicated in Figure 9 below.

### *US reprocessing plants at Hanford*

Two of the four reprocessing plants (see Figure 10 below) at the US Hanford Site, located in south-eastern Washington State were examined. The site, occupying an area of 1,450 $km^{2,}$ was established in 1943.

The chemical processing, waste management and related facilities are located in the area labeled 200 Areas East and West (see Figure 10). The two reprocessing plants in Area 200 West are enlarged and shown in Figures 11a and 11b.

The above two examples are a part of a number of sites studied. From the study of the images, the following "key" could be constructed for reprocessing plants:

- Large buildings that contains, the chemical treatment facilities as well as cooling ponds within the building or outside it;
- Often a reprocessing plant, that is used to produce weapons grade Pu-239, is located near the plutonium production reactor;
- A network of roads and railway line are associated with such a facility in order to bring spent fuel and take away waste products after reprocessing; and
- The waste product is either stored in large cylindrical or rectangular containers or even buried near the facility.

## 5. Some conclusions

From the examination of aerial and satellite-based high spatial resolution images detailed "keys" were derived for uranium enrichment and spent fuel reprocessing plants. These features are identifiable in satellite images. It is suggested now that using modest spatial resolution images (2.5m to 10m) computer-based, object-oriented image analysis should be carried to determine whether using the above "key" relevant sites could be identified automatically in a large area image. Once a site is detected, then they could be described using high spatial resolution images acquired over the sites of interest.

Figure 1. Maps showing Rokkasho area in detail.
Source: Based on http://cnic.jp/english/topics/cycle/rokkasho/rokkashodata#map

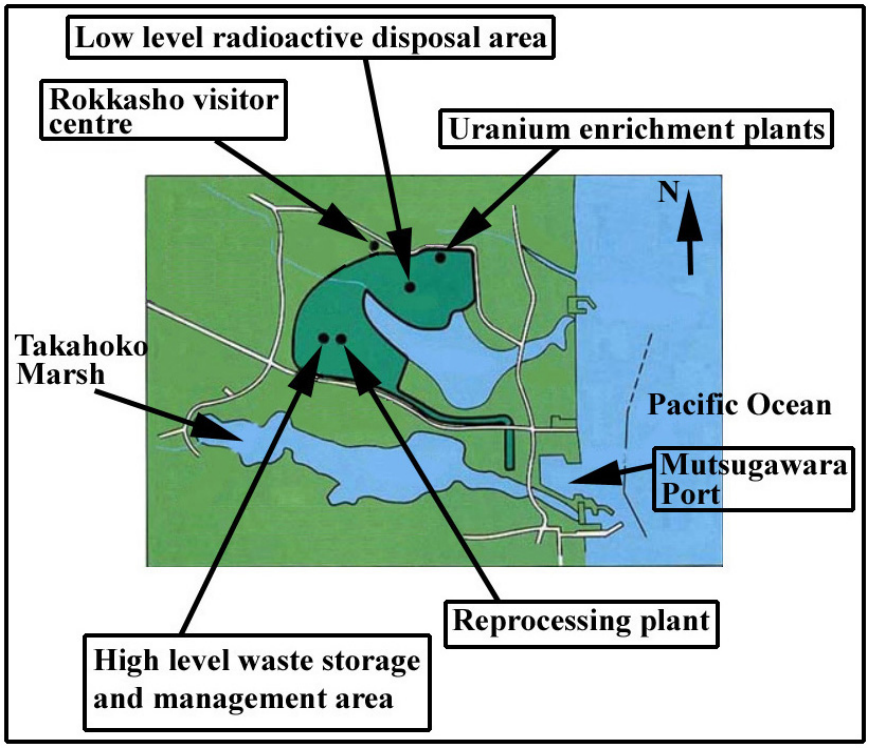


Figure 2. A Quickbird satellite image over the Rokkasho plant was acquired from the Google Earth sites over the Internet: http://maps.google.com/maps. Two enrichment cascade halls, the technical infrastructure building, the switchyard and the feed and the depleted $UF_6$ storage area are identified and shown as the enlarged sections. The roof mounted cooling system can be seen clearly in the satellite image over the two cascade halls.

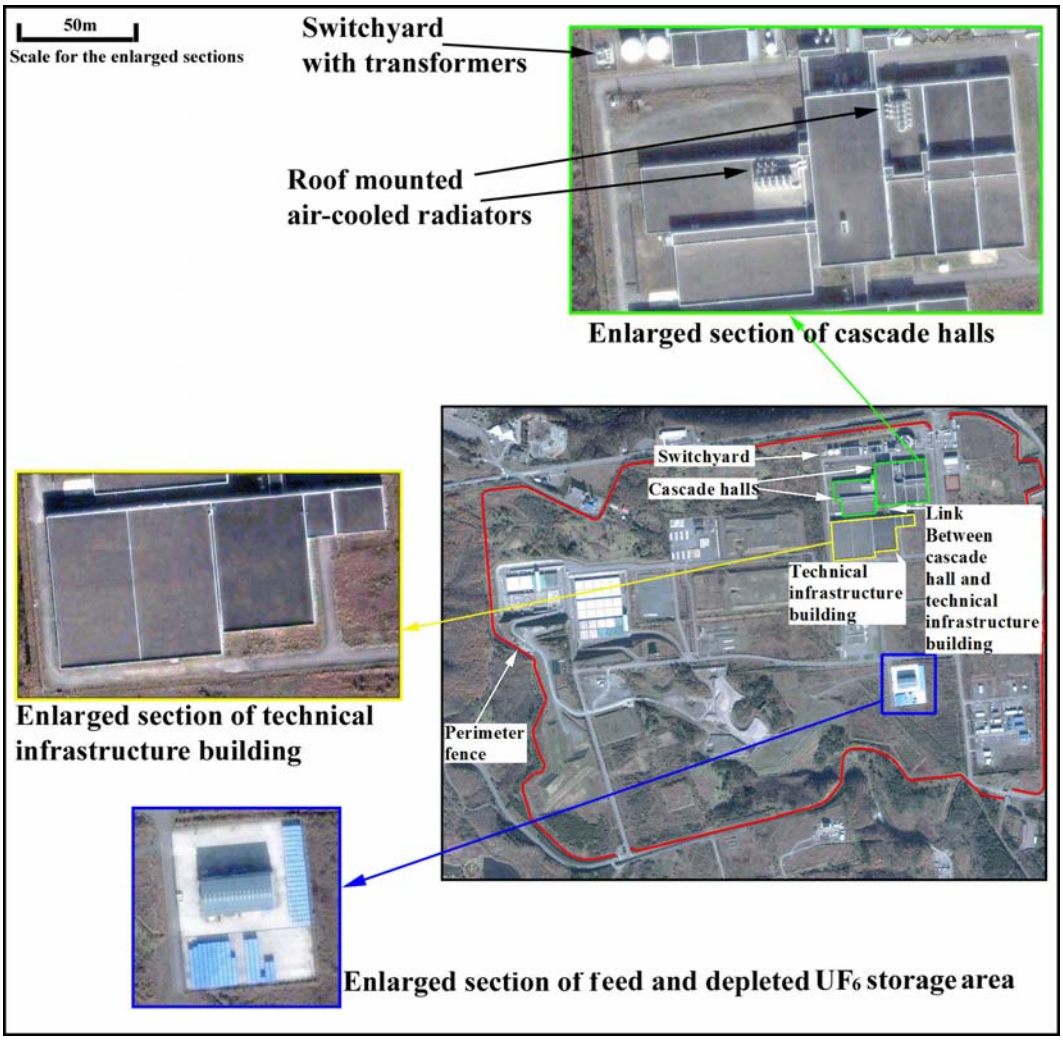


Figure 3. An over view of the town of Resende indicating the location of Brazil's Resende Gas Enrichment plant.
Source: http://maps.google.com/maps


Figure 4. This shows the Quickbird satellite image acquired over the Resende plant from the Google Earth sites over the Internet: http://maps.google.com/maps. Various key features are identified.

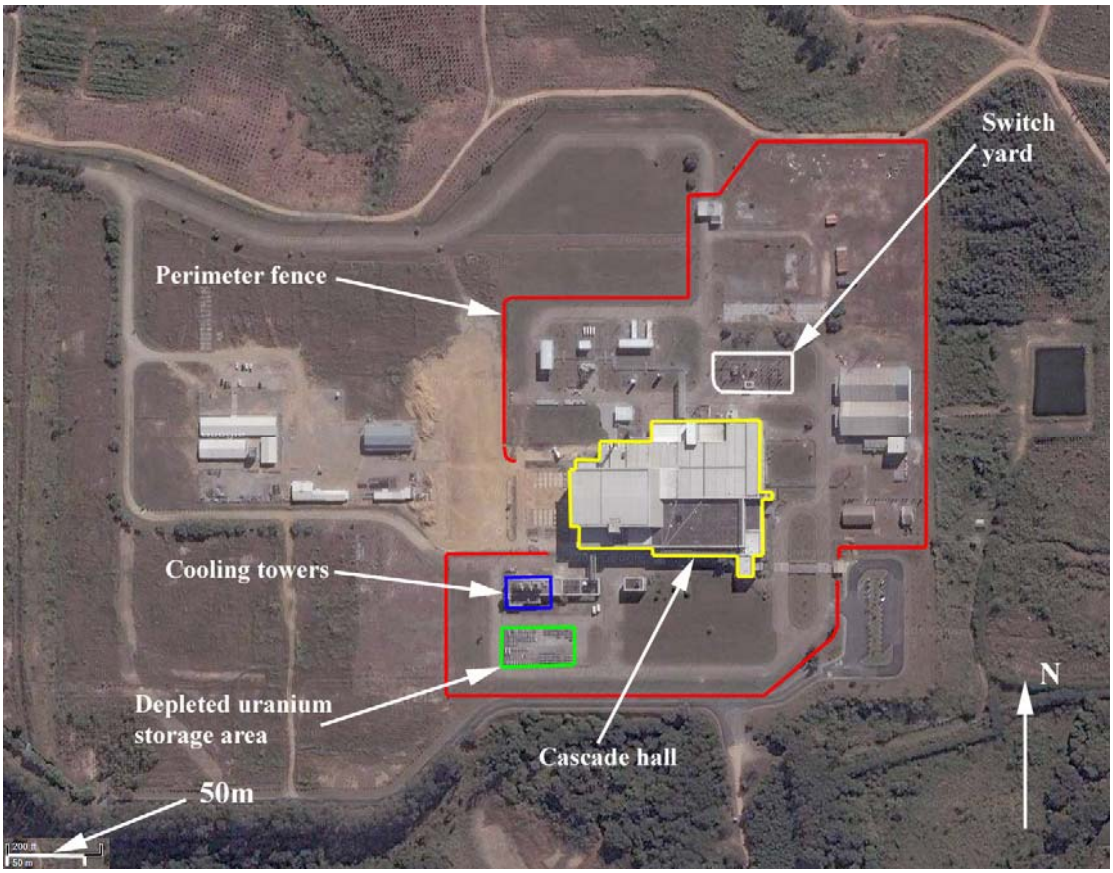


Figure 5. This is an overview of the US Paducah enrichment facility. On the image, acquired by probably the US Landsat satellite, the Paducah enrichment plant and the associated large conventional power plant can be identified.
Source: http://maps.google.com/maps

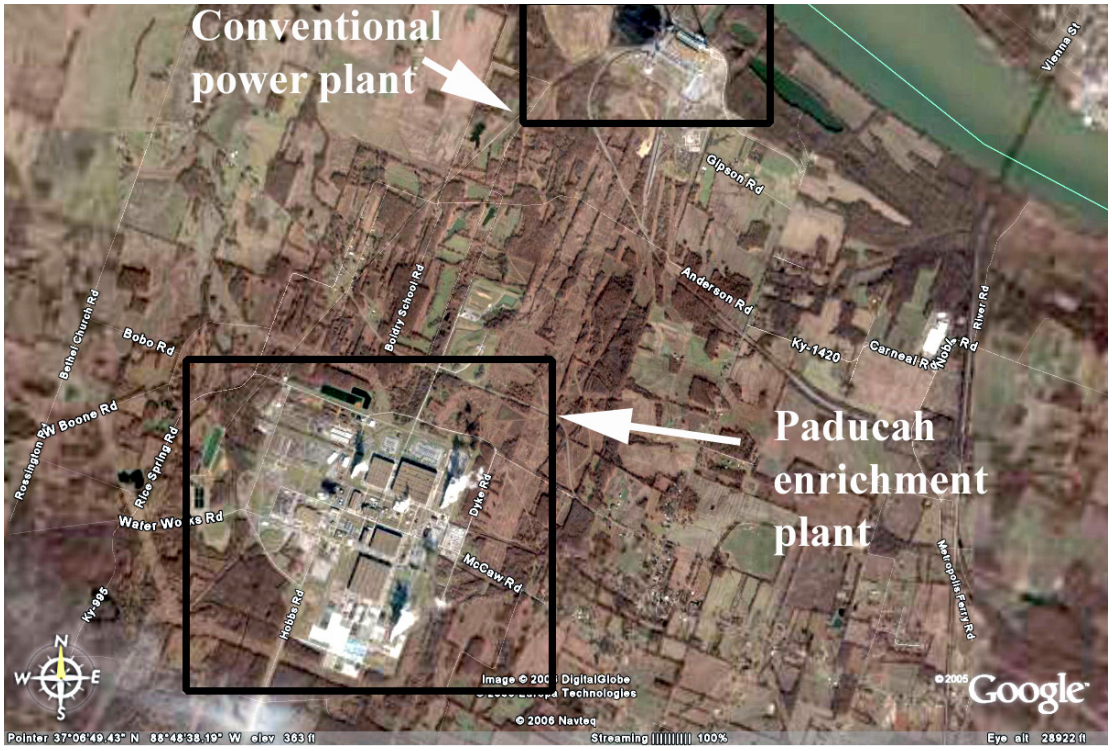


Figure 6. The conventional power plant near the US Paducah enrichment facility is enlarged from Figure 1. The barges carrying coal, the coal storage site, turbine and steam generators as well as the switchyard can clearly be seen in the image.

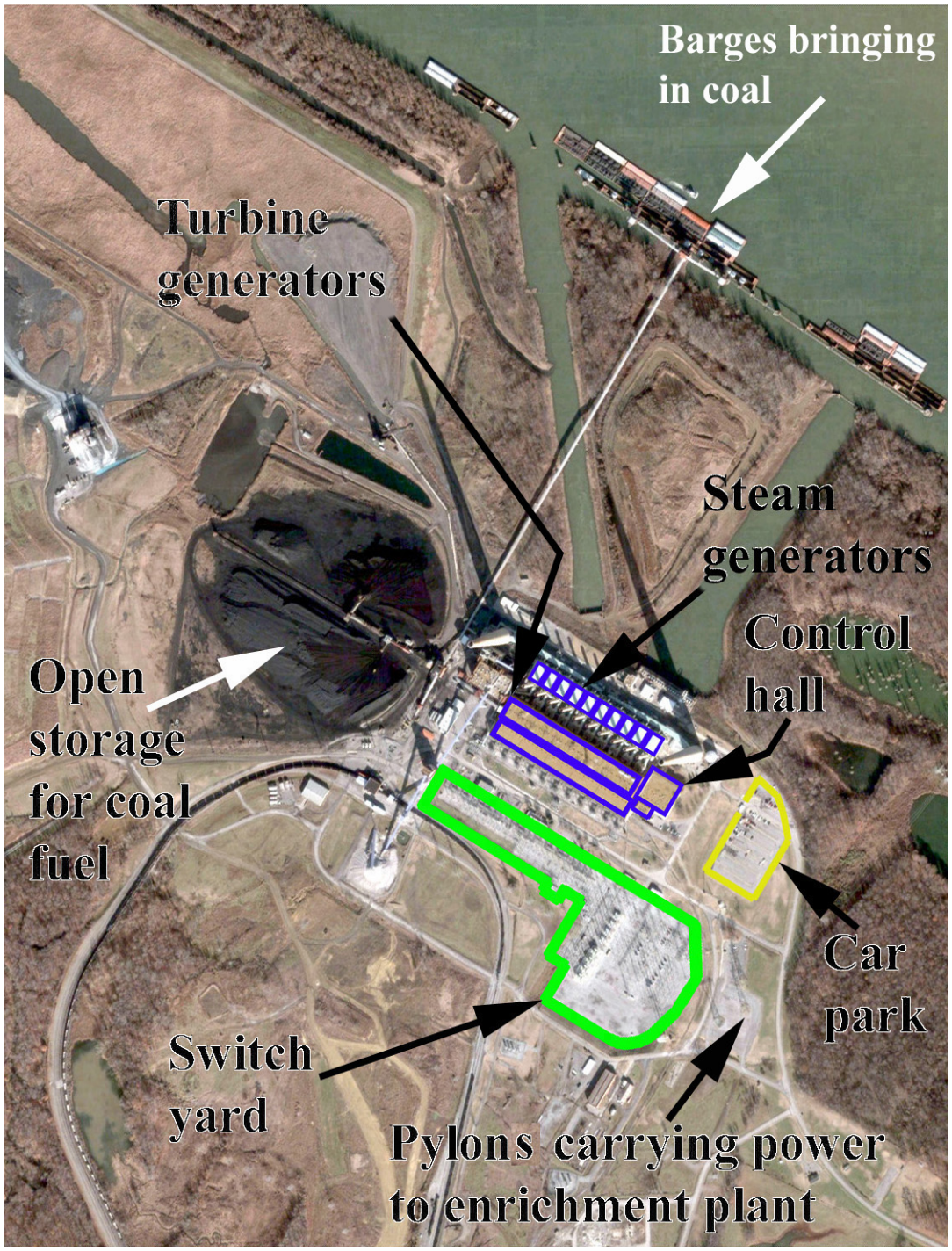


Figure 7. The details of the Paducah enrichment plant can be seen in this enlarged section from Figure 5. Various features such as the perimeter fence **(D),** possible storage area for the fuel caskets **(A)**, the switchyards **(B)**, the four enrichment halls **(C)**, the cooling towers **(E)** and possible feed plant **(F)** and possible waste burial area **(G)** can be identified.

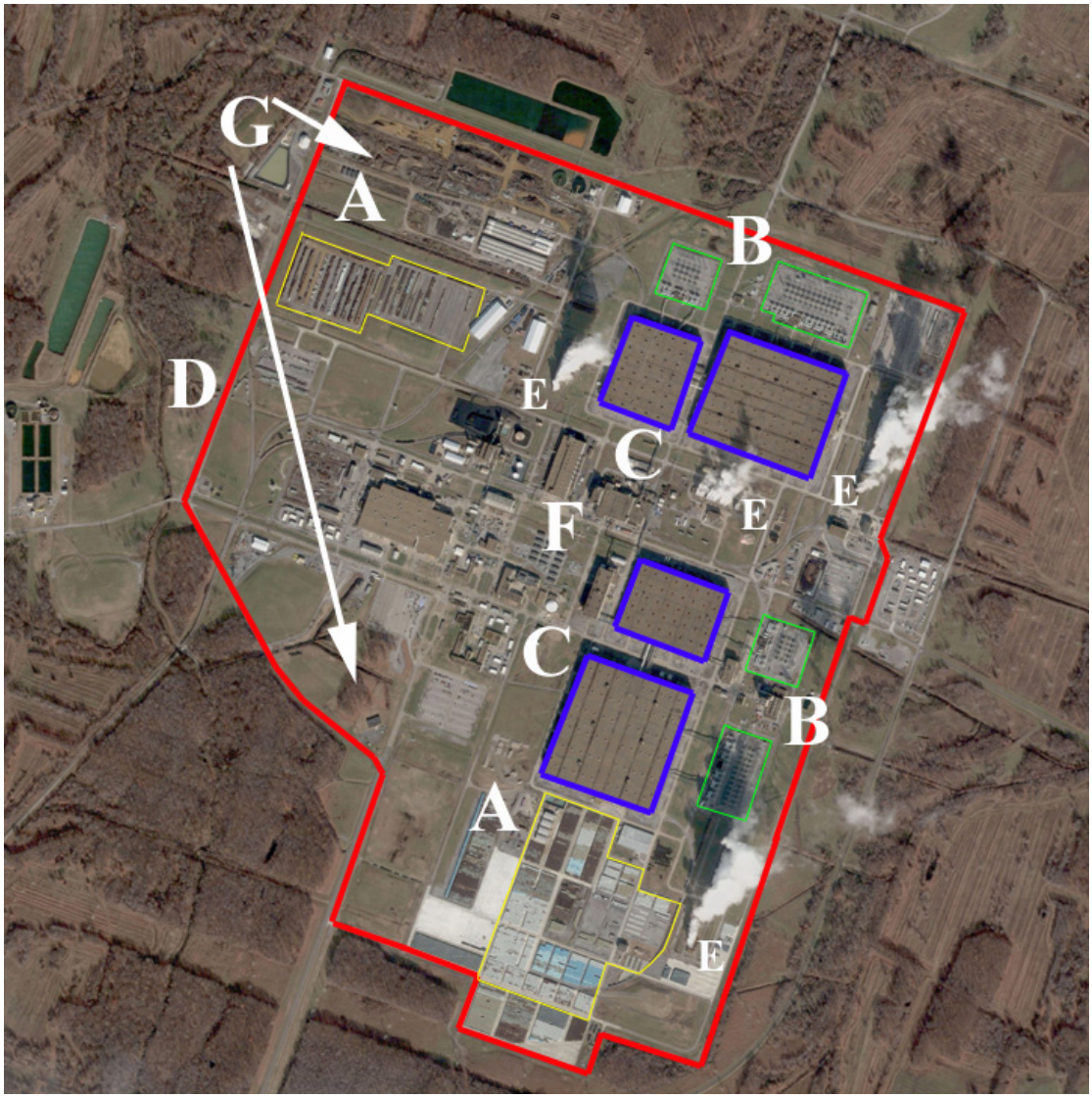


Figure 8. An aerial photograph of the THORP reprocessing plant in UK.
Source: The Thermal Oxide Reprocessing Plant - THORP, British Nuclear Fuels plc 1992.

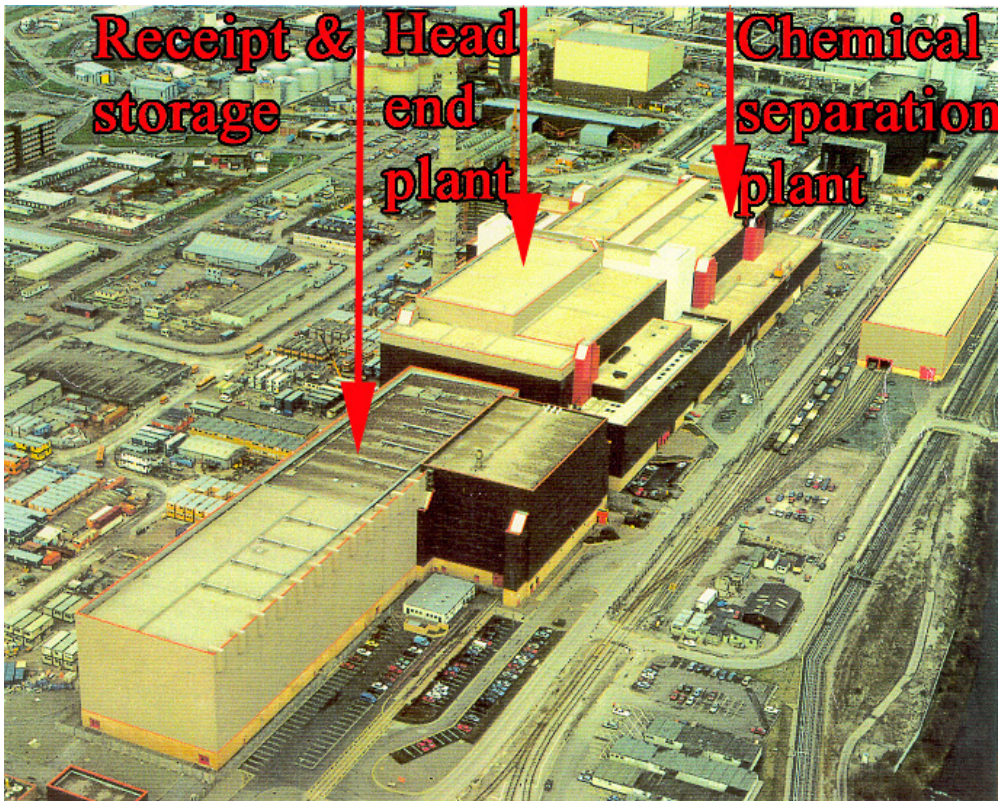


Figure 9. The Calder Hall reactors and the reprocessing plants can be identified. With the help of Figure 8, the three sections of the reprocessing

facility can be identified: **(1)** fuel reception and storage area; **(2)** head end plant where the fuel is cut up in small sections; and **(3)** chemical separation facility. Source: CNES/SPOT Image

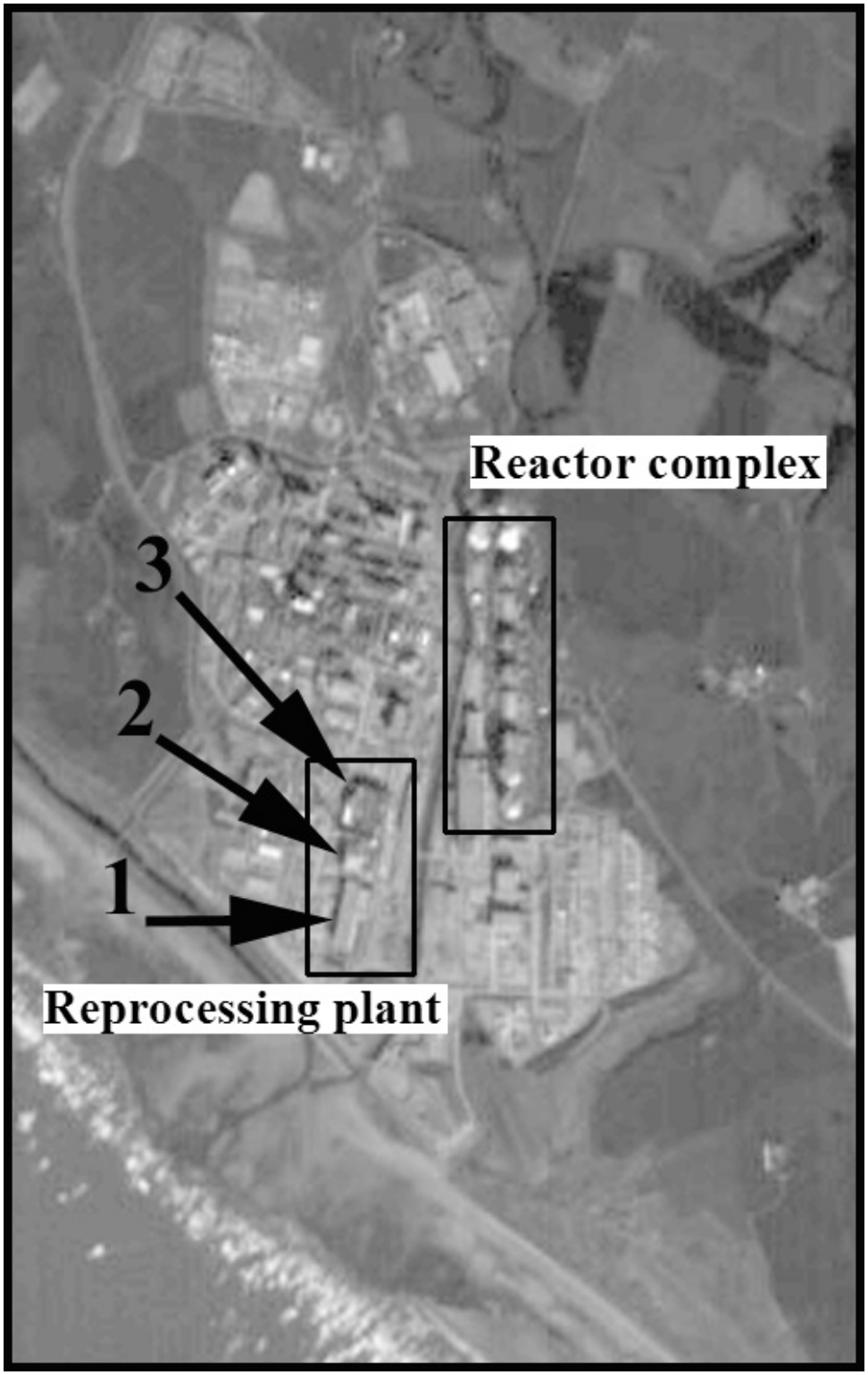


Figure 10. This a re-sampled extract of an image over the Hanford reactor and reprocessing facilities. Only an area where the reprocessing plants are located is shown. There are four possible plants, two in area 200 West and two in are 200 East.

Source: http://maps.google.com/maps


Figure 11a. Reprocessing plant 1 from Figure 10 is enlarged to show such details as waste containers, railway lines and roads. The long reprocessing plant building is un-mistakable.

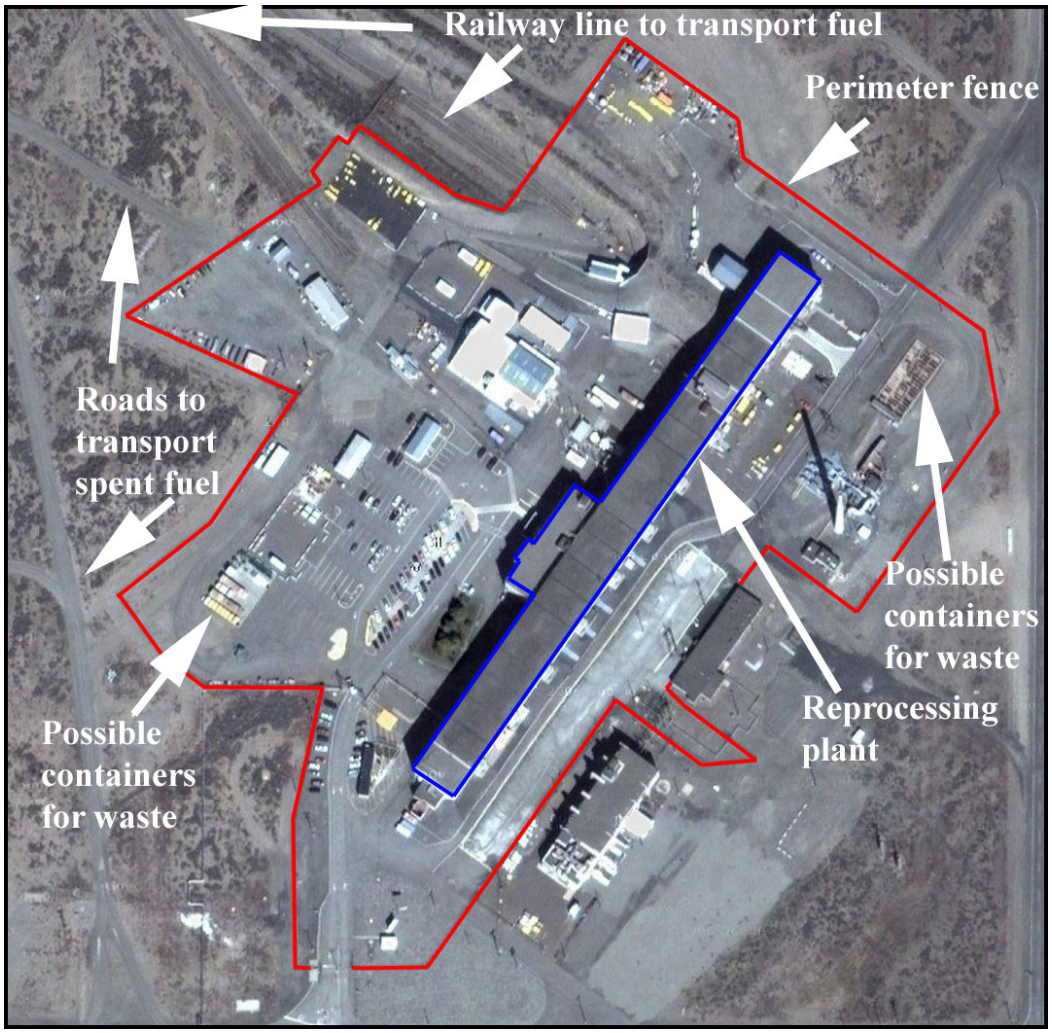


Figure 11b. Reprocessing plant 2 from Figure 10 is enlarged to show such details as large cylindrical tanks to store waste products, railway lines and roads. Again the long reprocessing plant building is un-mistakable. A pipe line connecting with other facilities in other parts of the complex can also be detected.


[*] This work was done in co-operation between the Research Centre, Jülich, Germany and the Department of War Studies, King's College London, UK. Views expressed in this paper are those of the author and should not be taken as necessarily representing official policy.

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[6] http://www.globalsecurity.org/wmd/facility/portsmouth_oh.htm

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[8] *Ibid* Jasani et al

# Session 15

# NDA III (Monte Carlo)

# Experiences, Trends and Challenges in Radiation Transport Simulations for Non Destructive Assay Systems

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***Abstract:***

*The Department of Safeguards of the International Atomic Energy Agency (IAEA) extensively uses Non Destructive Assay (NDA) systems to verify the nuclear materials declared by States with safeguards agreements. Effective implementation of NDA systems often requires the optimization of existing instruments and interpretation of the collected data requires accurate calibrations and sensitivity assessments. In both cases, the use of computerized simulation tools helps the IAEA to optimize and calibrate its NDA instruments.*

*Member State Support Programmes (MSSPs) have provided support in delivering the simulation codes. The general-purpose particle transport code MCNP developed by Los Alamos National Laboratory (LANL) is used for gamma and neutron simulations. MCNP-PTA, developed by JRC-ISPRA, is a modified version specifically devoted to the simulation of neutron coincidence counting instruments. A significant number of IAEA technical staff have demonstrated experience and skills in the use of these codes. In addition to their personal computers, they use a PC cluster with 16-cpu allowing minimum statistical uncertainty to be reached within short periods of time.*

*Significant experience has already been gained internally. Simulation work performed internally has been instrumental in resolving safeguards anomalies and in the effective implementation of nuclear material verifications. The main challenge now is related to the requirement to demonstrate objectivity in the validity and quality of the results gained through simulations. To this end and, with support from its MSSPs, the Department of Safeguards is developing specific internal procedures.*

*In addition to an overview of use of simulations for NDA instruments within the Department of Safeguards, the paper reports on the conclusions and recommendations of a recent IAEA workshop dedicated to the validation and quality management of the simulations.*

**Keywords:** Monte Carlo; Simulations; Calibration; best practice.

## 1 Introduction

The IAEA extensively uses a variety of NDA instruments to verify the nuclear material declared by States during safeguards inspections. Monte Carlo (MC) simulation codes are a powerful and reliable way to design and calibrate NDA instruments. They are widely used by instrument developers to optimize NDA instruments. Low cost computers like clusters of PCs allow complex simulations within short periods of time. In the past, it has always been critical to ensure the validation of the models and the methodology used for simulations. Such validation is particularly important when Monte Carlo methods are used to derive quantitative results, such as calibration parameters, or absolute detector responses. Validation of MC simulations is recognized as being sometimes more of an art than an exact analytical technique, and a lack of formal procedures often results in a failure to ensure the validation incurs reasonable costs. However, experienced users have developed their own practices to build confidence in their results. In 2006, the IAEA convened a technical meeting to stimulate

communication within the community of users dealing with NDA simulations and to capture their experience.

This paper describes several examples of the IAEA Department of Safeguards' experiences and achievements in using MC simulations and reports on the major recommendations formulated by experts on the validation and quality of MC simulation results.

## 2 Recent Experiences

The Division of Technical Support (SGTS) within the IAEA Department of Safeguards has established an in-house capability to perform MC simulations in order to address verification needs in a timely manner. In particular, MC simulations are used on a daily basis to improve and customize existing instruments or to calibrate NDA systems in order to address the verification of nuclear materials when experimental calibrations either do not exist or are difficult to perform. A group of experienced users now conducts simulations using a compact cluster of 16 AMD 64-bit Athlon water-cooled CPUs running Linux. The cluster has been available to users for the past year via the IAEA LAN and offers a total calculation speed equivalent to around 40 GHz.

In the last three years MC simulations have been applied to various kinds of instruments as illustrated by the few examples described in the following sections.

### 2.1 Calibration for low-resolution gamma spectrometry

#### 2.1.1 Enrichment verification of fresh fuel pellets

NaI(Tl) detectors associated with IMCA, (IMCN) are used to perform enrichment verification on CANDU fresh fuel pellets. A dedicated MS Windows application was created to support the establishment of the calibration of the NDA equipment utilizing automated MCNP calculations. The application developed expands on previous work [1] and takes into account the benchmark of the numerical calculation against empirical measurements of reference material. It also supports the analysis of experimental spectra to compare measured and declared enrichments. Figure 1 shows an example of the MCNP model generated by the code along with the input window of the modelled parameters.

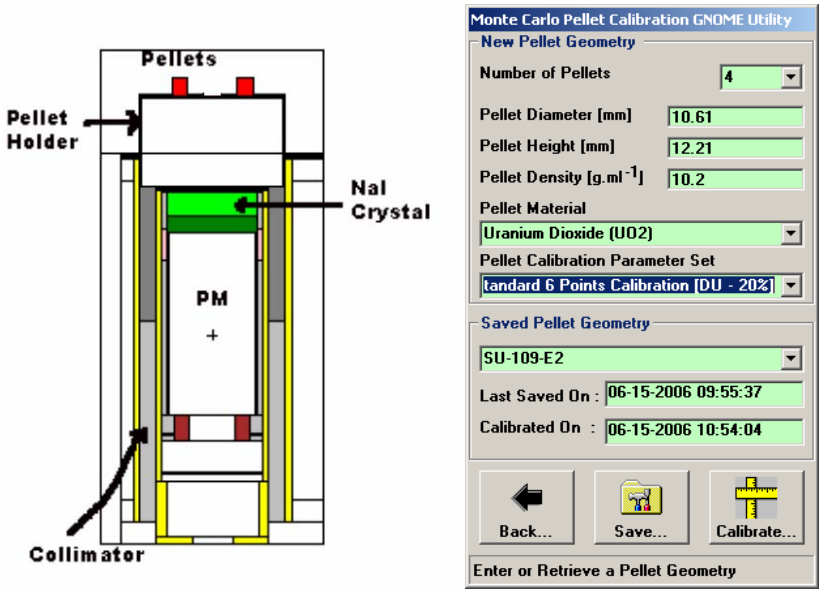


*Figure 1: Cross section of a MCNP model (LHS) obtained from the GNOME calibration software (RHS).*

Implementation of the MC calibrations for NaI(Tl) detectors led to the successful verification of CANDU fresh fuel pellets. The observed difference between calculated and declared enrichment over 24 measurements (spanning 13 types of fuel pellets with various geometries and densities) returned a 4.62-% standard deviation, with a mean difference of 0.97%. This is shown graphically in Figure 2.

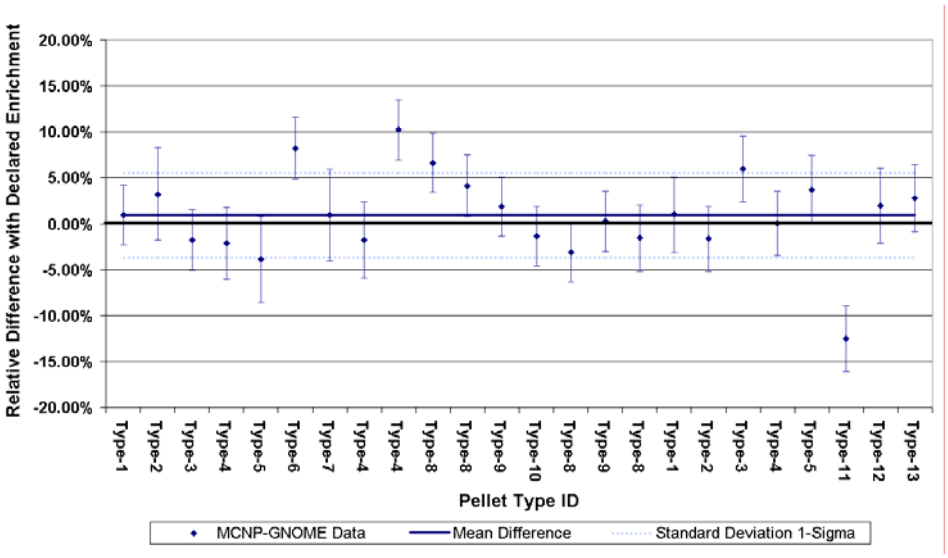


*Figure 2: Observed bias between MCNP derived and declared enrichments*

### 2.1.2 Verification of fresh fuel for experimental reactors

The same instrumentation based on NaI(Tl) detectors is used for the verification of research reactor fresh fuel elements. The large number of variations of design of those fuel elements led to the generation of a versatile tool able to generate MCNP input files according to the exact fuel design. The MS Windows application MC-MTR supports both calibration calculations and the derivation of a benchmark factor based on experimental characterization of the detector response. Figure 3 shows the MC-MTR application interface.

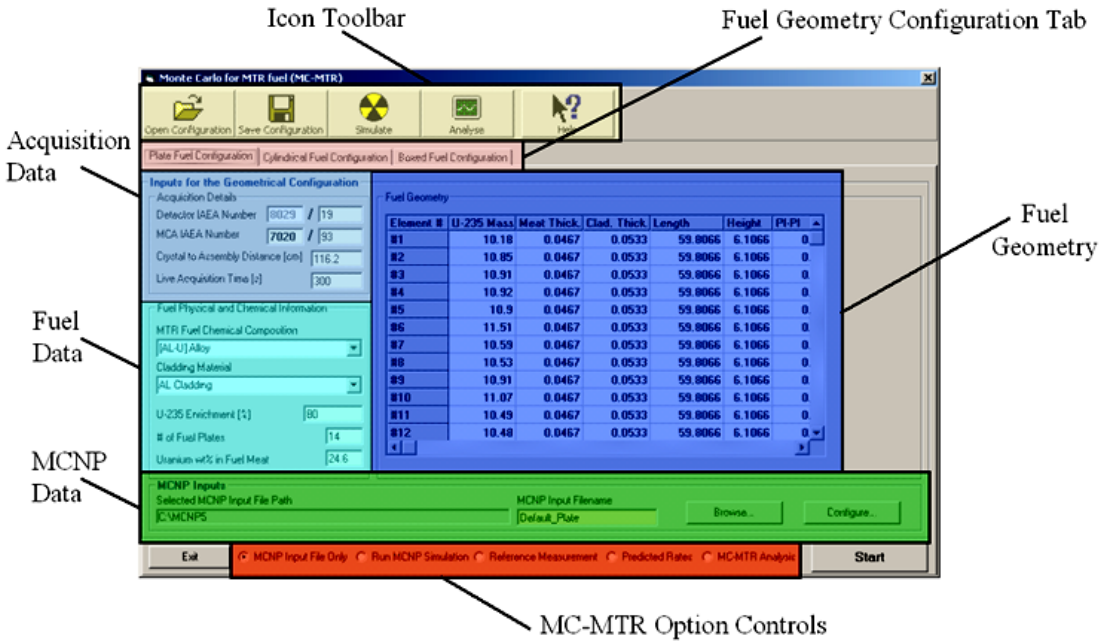


*Figure 3: Windows interface of application for calibration of NaI(Tl) detector for verification of experimental reactor fresh fuels*

Successful implementation of the MC-MTR application strengthened the IAEA's capability to verify an extended range of experimental reactor fresh fuel with improved accuracy.

## 2.2 Calibration of neutron instruments

The IAEA operates a large diversity of neutron coincidence counting instruments like the AWCC or UNCL, which have actual calibrations covering a limited range of measurements. Assaying unusual

nuclear material like large canisters of uranium powder or LWR fresh fuel assemblies with the presence of burnable poisons requires establishing valid calibration. The IAEA now generates such calibrations by means of MC simulations.

### 2.2.1 Calibration of a customized AWCC for uranium verification

Safeguards implementation at a fuel fabrication plant operating HEU down blending requires quantitative verification of feed and product materials contained in specific containers for which experimental calibrations are not practical. On the basis of a baseline MCNP model of a standard AWCC, which was provided by a MSSP, the IAEA has developed a customised model to perform these calibrations. A cross section of the model is shown in Figure 4.

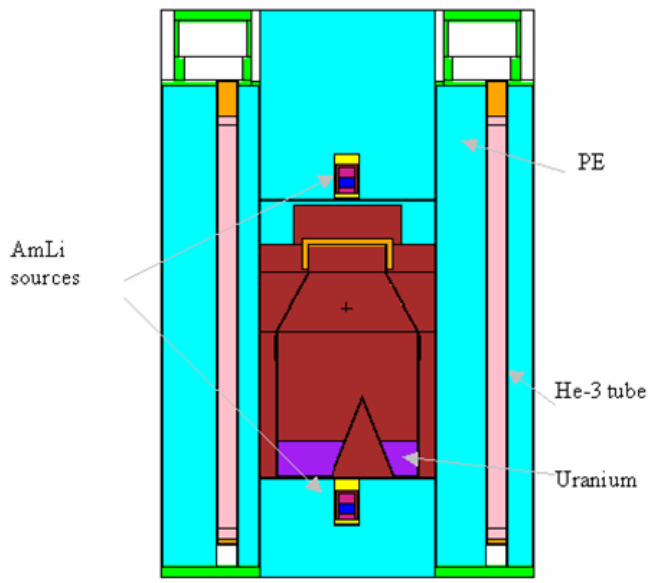


*Figure 4: Vertical cross section of the customized AWCC with uranium canister*

The calibrations generated by MCNP-PTA simulations have been used by the IAEA to perform 3000 measurements supporting close verification of the down blending process.

### 2.2.2 Calibration of a customized UNCL for LWR fresh fuel with burnable poison

Increasing initial enrichment of LWR fresh fuel is a general trend in the nuclear industry. High enrichments go together with the use of burnable poisons under various forms, such as specific rod or inclusion of neutron absorbing material within fuel rods. The nature, concentration and location of the burnable poison within the fuel assembly impact the response of the UNCL, which therefore needs to be specifically determined. With support from a MSSP, the IAEA developed MCNP models of its UNCL instruments for PWR, BWR and VVER-1000 fuel assemblies, as illustrated in Figure 5. The IAEA is now able to determine accurate calibration parameters that can be used to assay any kind and design of LWR fresh fuel. In addition, correction factors to the detector response can be calculated with MC modelling for fuel designs with current calibrations to account for such things as: varied burnable poison rod positions and burnable poisons other than Gadolinium.

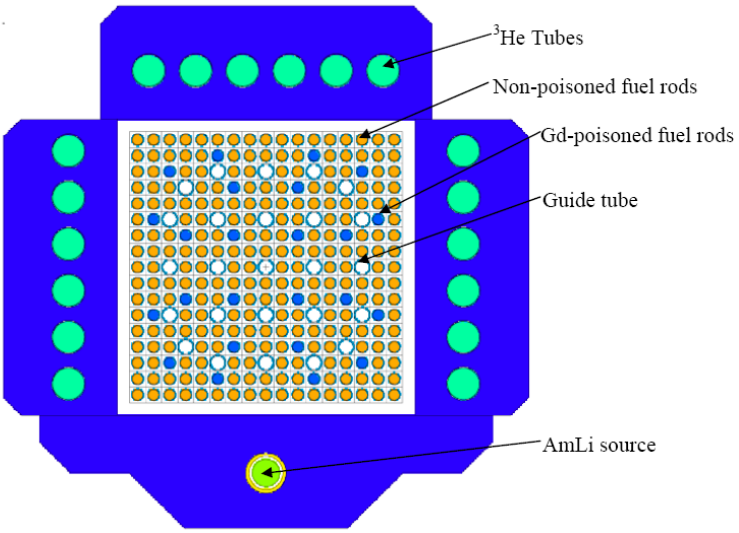


*Figure 5: Horizontal cross section of the UNCL model for the verification of PWR 17x17 fresh fuel with gadolinium rods*

### 2.3 Design and optimisation of instruments

SGTS uses MC simulations on a daily basis to investigate the appropriateness of NDA solutions addressing a large variety of configurations and diversion scenarios as requested by the IAEA Divisions of Operations. The principal benefit is the timeliness of the response. Confidentiality issues are also important, as the design information placed under IAEA custody within the Design Information Questionnaires cannot be disclosed to third part entities. Furthermore, evaluation of specific diversion scenarios should certainly not be advertised.

However, the IAEA still relies on industrial suppliers and MSSPs for the detailed design of new instruments.

## 3 Trends and challenges

Although very effective, MC simulations are time-consuming activities and the IAEA has only limited resources to devote to this activity. As a result, efforts are being made to strengthen the effectiveness of the simulation work by the team of experienced MCNP users. The ESARDA NDA working group with IAEA participation represents another team effort which is presently developing an internationally recognized best practice guide for simulations related to NDA instruments.

To address the important issue of ensuring the validation and quality of the results gained through MC simulations, the IAEA organized a technical meeting in December 2006. The meeting gathered 14 experts from nine Member States and four IAEA experts. It resulted in the formulation of ‘best practice guidelines’ for the application of Monte Carlo methods to safeguards problems. The experts’ recommendations are now in the process of being applied to support the establishment of internal IAEA procedures consistent with these best practice guidelines. The following sections report on the recommendations of the expert group in the areas of:

- Formalization of Monte Carlo Projects
- Evaluation and minimization of errors attached to simulations;
- Definition of experimental benchmarking for the validation of models;
- Effective Peer Review;
- Elaborating calculation plans and Quality management system of the simulations.

### 3.1 Formalization of Monte Carlo projects

The first recommendation expressed by the experts was that any work involving simulations should be handled as a project, which implies that customers, end-users, suppliers, contributors and reviewers are identified and work on the basis of user requirements developed and agreed upon by the customers and suppliers. Of course, such organization should be flexible enough to be applicable to small projects while structured enough to properly address more complex projects. The major steps of a Monte Carlo Simulation Project were identified as follows:

1. *User requirement:*
   a. *Definition of needs, scope, constraints and expected results;*
   b. *Collecting information to model;*
   c. *Responsibilities.*
2. *Modelling:*
   a. *Baseline model;*
   b. *Sub modelling for benchmarking purposes;*
   c. *Production models.*
3. *Experimental benchmark:*
   a. *Define the experimental measurements;*
   b. *Perform the measurements.*
4. *Peer review:*
   a. *Develop peer review plan;*
   b. *Perform and report peer review.*
5. *Develop calculation plans specifying the calculation cases to be run;*
6. *Perform the calculation using batching script or special user interface according to calculation plan:*
7. *Evaluate the calculation results:*
   a. *Extract and post process the results;*
   b. *Evaluate uncertainties.*
8. *Report all steps.*

### 3.2 Evaluation and minimization of errors attached to simulations

Evaluating the accuracy of MC simulations goes far beyond the evaluation of statistical errors, which are properly evaluated by the codes. As a general rule, the requested overall level of uncertainty required should match the efforts used to control and understand errors. A good understanding of the problem is a key factor to accurately model the problem. Such parameters as detector geometry or material composition and density in the vicinity of the detector must be adequately understood. Incomplete knowledge of the instrument design, and even more frequently, of the sample characteristics are the main limiting factors objectively impacting upon the uncertainty. Sensitivity studies associated with the model are the most convenient way to improve the understanding of the problem and therefore to limit and estimate the uncertainties.

For small projects, a fast and cheap uncertainty evaluation can be obtained by calculating lower and upper limits by changing all parameters at once.

### 3.3 Definition of experimental benchmarking for the validation of models

Experimental benchmarking is a must for any MC simulation aiming at estimating any kind of absolute figures. Reference experimental measurements of appropriate point sources carried out in well controlled and documented conditions are key to successful results. Although useful, benchmarking with non-representative standards represents more of a confidence building measure than a true validation exercise. The quality of the characterization (including absolute emission rate) of the point sources is critical as it directly impacts upon the uncertainty of the experimental benchmark. For neutron coincidence counting instruments, problems experienced with $^{252}$Cf sources (i.e. unknown amounts of the longer lived $^{250}$Cf in the source) may be overcome through the selection of other isotopes like $^{248}$Cm.

### 3.4 Effective Peer Review

Validating simulation results and assessing their uncertainty is often based on many empirical factors and a traditional procedure is not the optimal and transparent way to ensure the overall quality of the simulation work at reasonable cost.

There is a consensus that peer review provides an efficient means for increased assurance in validity of the MC model and results. A peer review should be undertaken as part of a project. Within the project plan the peer review should be specified as consistent with the overall objective of the project and with available resources. In order to facilitate and ensure the effectiveness of the peer review, it must be conducted independently from the developers but in a transparent manner. It may involve non-MC experts who have an understanding of the physics of the problem. Confidentiality issues must be taken into account to organize external peer reviewing.

### 3.5 Elaborating calculation plans and quality management system of the simulations

Once the validation of the model and the methodology has been achieved, the production phase should be effective and traceable. Establishing calculation plans implemented by the use of batching scripts is highly recommended. They can automatically merge modular input files, initiate calculations and retrieve the raw results, thus allowing effective use of the calculation power and traceability of the calculations. When economical, specific graphical interfaces may be useful as they allow safe use of MC calculations by non-experts.

Proper reporting is critical and the documentation provided with the results should be comprehensive. In addition to the results, documentation should contain a description of the models and their validation, the calibration and experimental benchmark report when used and the peer review report. It is recommended that results expressed in end-user units are always supplemented by raw results from calculations.

## 4 Conclusions

During the previous years, the Department of Safeguards has been establishing an in-house capability to conduct Monte Carlo (MC) simulations of NDA systems. MC simulations are conducted on a daily basis by a group of experienced MCNP users to provide support to the IAEA Divisions of Operations in performing effective verification of a variety of nuclear material with optimized instruments and accurate calibrations. Despite possessing some independent capabilities, the IAEA still relies on the support of MSSPs in the MC simulation area to provide baseline models of existing instruments or in the course of the development of new instruments. The key issue of validation and quality assurance attached to the simulation results has been addressed and a technical meeting of Member States' experts was organized in Vienna in December 2006. The recommendations formulated by this expert meeting will support the establishment of formal internal procedures where applicable. In the meantime the IAEA continues to participate in the ESARDA NDA working group where it delivers feedback of its experience in MC simulation matters as a contribution to the establishment of an internationally recognized best practice guide for MC simulation related to NDA instruments.

## 5 References

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# Simulations of Neutron Multiplicity Counters with MCNP-PTA.

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***Abstract:***

*The paper is about the improvements and new features related to simulation of neutron multiplicity counters with the MCNP-PTA (Pulse Train Analysis) code developed at JRC Ispra. Simulation models of multiplicity counting electronics, like the JSR-14, AMSR-150, and MI-PTA have been added, paying extra attention to model faithfully all of their characteristics (de-randomising input buffer, pulse pair resolution, clock frequency and direction in time of the gates). Moreover simulation models of several de-randomising mixer circuits have also been added to the PTA program. This is especially important for the correct simulation of measurements with large Pu samples, which involve very high-count rates. In the case of multiplicity counting accurate modelling is mandatory, as will be shown for an Active Well Coincidence Counter. Attention will also be given to the importance and limitations of the nuclear data used in the simulation.*

Keywords: cluster, PVM, MCNP, MCNP-PTA

## 1. Introduction

The MCNP-PTA code started as a tool for the modelling of Neutron Coincidence Collars: active interrogation of BWR and PWR fuel assemblies [1]. Over the years the excellent agreement between simulation and experimental results, established confidence in the Monte Carlo (MC) modelling of the NCC instrument, and finally led towards the acceptance of MC techniques for the calibration of neutron counters where particular nuclear reference materials were unavailable [2].

Extension to neutron multiplicity counting of Pu samples was the next step forward in the evolution of the code. What follows is a summary of the various aspects towards the simulation of multiplicity counting with MCNP-PTA.

## 2. Pulse train scanning direction

Faithful simulation of all steps in neutron transport and pulse train analysis electronics is one of the goals of the MCNP-PTA code. It came thus as a surprise that the first simulation results for the triples rates, as reported for instance in the ESARDA Multiplicity Benchmark Exercise [3] were systematically too high.

The same systematic error was observed between experimental results obtained with new virtual instrument electronics and those obtained with an AMSR 150, when measuring Pu samples with the Scrap Neutron Multiplicity Counter (SNMC) at the Perla laboratory. The Virtual Instrument (VI) electronics relies completely on a PC for the pulse train analysis. Today, PCs are fast enough to run multiplicity counting software in real time. One of the advantages is the flexibility of the VI. For large Pu samples JSR-14 and AMSR electronics report an error for the gate multiplicity exceeding the maximum of 255. In the case of the VI the modification of the

analysis software was straight forward, and the maximum gate multiplicity was adjusted. The pulse train analysis algorithms are the same for the VI and for the PTA code, and were checked for programming errors that could explain the systematic error. Eventually it was found that the VI and the PTA code were both scanning backwards in time, as is suggested by the schematic diagrams of standard shift register electronics like for instance the one by Swansen et al [4].

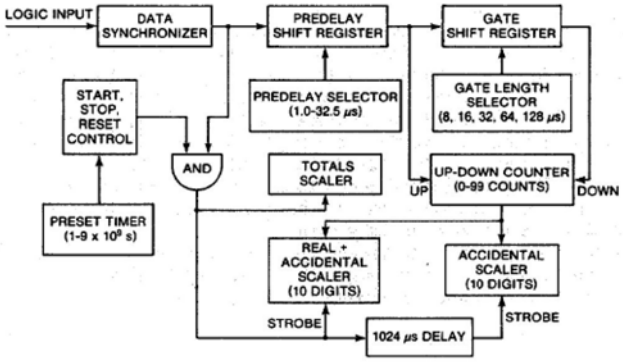


**Figure 1 Schematic diagram of standard shift register electronics [4].**

The AMSR and JSR-14 in the contrary scan both forward in time, as it is the case for Hage's point model equations for multiplicity counting. While results for the singles and doubles are independent of the direction in time of the scanning, this is not the case for the triples as is illustrated in figure 2.

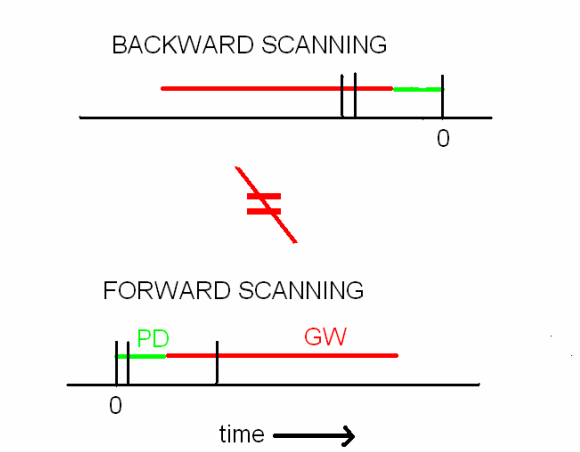


**Figure 2 The direction in time used in the scanning algorithm is important for the triples. When forward scanning is applied to the above segment of a pulse train, we obtain the following results: M[0]=1 and M[1]=2. When analysing the same pulses backward in time the results are: M[0]=2 and M[2]=1.**

After reversing the scanning direction, the systematic error disappeared. As mentioned earlier, backward scanning gives a larger triples rate, and thus the relative statistical error should be also smaller for the same measurement duration. In order to apply this advantage in multiplicity counting, one should find the equivalent of Hage's point model equations for the case of backward analysis. The new version of the PTA (Pulse Train Analysis) code provides both forward and backward scanning of pulse-trains. In order to test MCNP-PTA simulation results with point model calculations the IDEAL analyser has been introduced. This analyser has both zero dead-time, pulse pair resolution and clock interval. When pulse trains generated with the Point Model Pulse Generator (PMPG) program are analysed with the PTA program, the agreement is excellent (doubles $\leq$ 0.12 %, triples $\leq$ 0.4 %).

## 3. Triples modelling

Accurate modelling of the detector instrument is mandatory to obtain reliable simulation results. The point model equation for the singles count rate of a detector only depends on a relatively small number of physical quantities: the neutron source strength $F_s$, the neutron multiplication M, the alpha ratio $\alpha$, the average number of neutrons per spontaneous fission $<\nu>$ and the detection efficiency $\varepsilon$. Generally, the aim of MC simulations is to calculate count rates as a function of the actual masses of a number of isotopes in a sample. Uncertainties in the half-lives of the isotopes introduce a first error in the MC simulations, since one has to calculate the isotopic composition at the measurement date. Moreover, calculation of the source strengths depends on the decay constants and in the case of the $\alpha$, also on the thick target yields for $(\alpha,n)$ on oxygen. Finally, a further source of error from source related nuclear data is the uncertainty in $<\nu>$.

Accuracy of the result also depends on the modelling of neutron transport, which in the case of the singles rate is represented by M and $\varepsilon$. In terms of the MC calculation, these are most influenced by the homogeneity and density of the sample, the spectra of neutron emerging from Spontaneous Fission (SF), Induced Fission (IF) and the Alpha-Neutron reactions (AN), and of course the dimensions, densities, composition and cross-sections of the materials making up the detector system.

Of major importance are the correct description of the sample, the amount and spatial distribution of the Polyethylene moderator and the amount and spatial distribution of the $^3$He counter gas (not only in the active zones of the counter tubes, but also in the dead zones: $^3$He is very effective neutron absorber). The modelling of other materials is often of lesser importance, Cadmium is of course an exception to this rule.

Things are getting more difficult when calculating doubles rates, mainly because of the quadratic dependencies, $\varepsilon^2$, $M^2$ and $<\nu(\nu-1)>$, and for the introduction of time dependency in the simulations. Time dependency is related to the doubles gate fraction $f_D$, which in turn depends on the neutron die-away time constant $\tau$.

The real challenge for MC modelling is the calculation of the triples rates. Quadratic and cubic dependencies are the norm: $\varepsilon^3$, $M^3$ , $(f_D)^2$ and $<\nu(\nu-1)\ (\nu-2)>$. These non-linear dependencies put a heavy strain on the MC model, what might look of lesser importance to a modeller, may in fact become important at closer examination.

For instance the density of PolyEthylene (PE), which according to data sheets may vary from 0.92 to 0.97 g.cm$^{-3}$ has a limited influence on the neutron detection efficiency as is illustrated in figure 3.

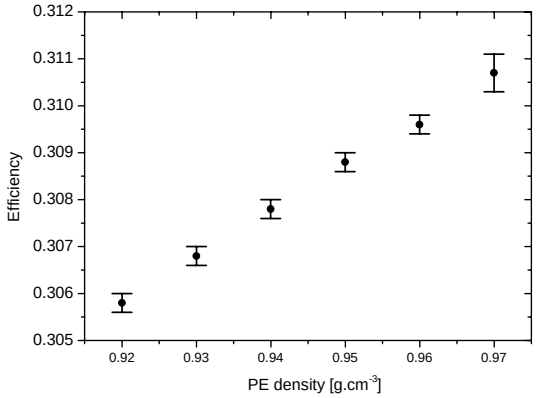


**Figure 3 MCNP simulation results for neutron detection efficiency as a function of Polyethylene (PE) density for a $^{252}$Cf source placed in an AWCC.**

Although one may be tempted to fine tune his MC model to experimental results by adjusting the PE density, it is recommended instead to measure directly the density. For instance for the top and bottom plugs of an AWCC located in the PERLA laboratory (JRC Ispra) a density of 0.952 ± 0.004 g.cm$^{-3}$ was found. In order to measure the density large metal parts (disks, handles) were removed, the mass was measured with a precision scale and the volume estimated from dimension measures with a callipers. PE density may depend on manufacturer and even production batches, nevertheless measured values (from non destructive analysis) for various neutron counters were all close to 0.95 g.cm$^{-3}$. It appears that neutron counting instruments are commonly made of High Density PolyEthylene (HDPE), whose density ranges from 0.95 to 0.97 g.cm$^{-3}$. For future designs we would advise to request from the manufacturer a small material sample taken the same production batch as that used for the construction of the neutron counter. This would enable further analysis of material structure in the future, like the eventual presence polycrystalline structures.

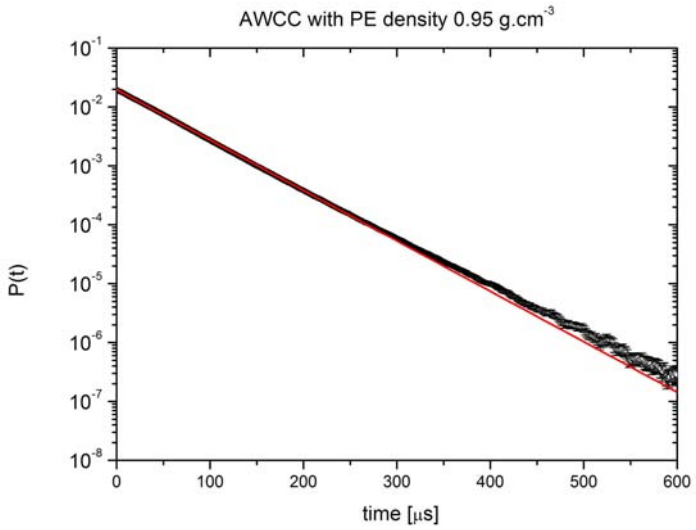


**Figure 4 MCNP simulation results of neutron die-away for a $^{252}$Cf source placed in an AWCC with a PE density of 0.95 g.cm$^{-3}$. Straight line is fit to single exponential decay function.**

The effect of the PE density is not only limited to the neutron detection efficiency. Whit MCNP the neutron die-away curves were calculated for several values of PE density (see figure 4) and fitted to single (or double) exponential decay functions. The combined effect of the dependency on PE density of the detection efficiency and die-away, shown in figure 5, results in a much larger variation of the triples rate than for the singles rate.

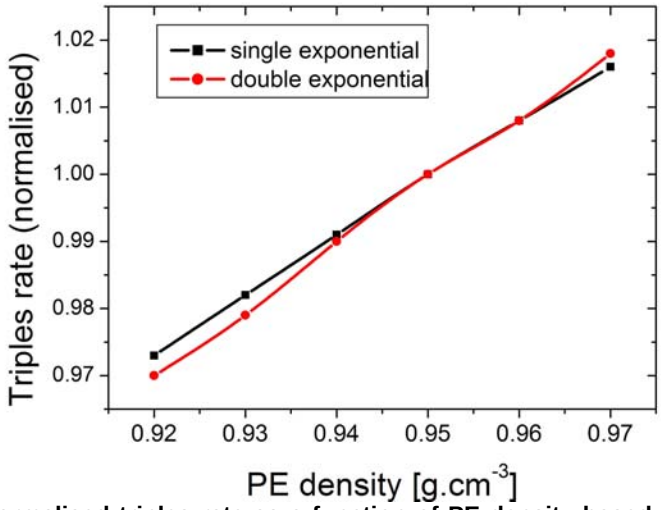


**Figure 5 Normalised triples rate as a function of PE density based on MCNP simulations and fitting with single or double exponential decay functions.**

Another example of increased sensitivity of the triples rate on modelling accuracy of is represented by the wall thickness of the counter tubes. The AWCC has 42 counter tubes with aluminium walls. The external dimensions of the tubes are specified by the tube manufacturer and can be easily verified by direct measurement. This is not the case for the inner dimensions of the tube, which are calculated assuming a certain wall thickness. For instance, in the model that was used to calculate the pulse trains for the ESARDA Multiplicity Benchmark Exercise [2], a wall thickness of 0.4 mm was used. Such a wall thickness corresponds more to that of counter tubes with stainless steel walls. In fact aluminium is softer than steel, it has a much smaller Young's modulus, and thus thicker walls are necessary to withstand the pressure (5.4 atm) of the counter gas: values of 0.081 to 0.089 mm (0.032" to 0.035") are common according to [5]. Again the effect on the singles is marginal: when using the thicker walls the value is decreased by 1.6%. However, the combined effect of die-away time constant and efficiency result in much larger reductions of doubles and triples, 4.8% and 7.9% respectively.

**Table 1 MCNP simulation results for singles, doubles and triples rates for old and new values of counter tube wall thickness. Fit results of neutron die-away time constant and the corresponding doubles gate fractions are also given.**

| Case | wall thickness | | singles | die-away | | doubles | triples |
|---|---|---|---|---|---|---|---|
| | [inch] | [cm] | $\varepsilon$ | $\tau$ [$\mu$s] | $f_D$ | $\varepsilon^2 f_D$ | $\varepsilon^3 (f_D)^2$ |
| Old | 0.0157 | 0.040 | 0.3090 | 50.75 | 0.656 | 0.063 | 0.0127 |
| New | 0.0320 | 0.081 | 0.3042 | 52.64 | 0.646 | 0.060 | 0.0117 |
| Ratio (New/Old) | | | 0.984 | 1.037 | 0.985 | 0.952 | 0.921 |

As mentioned earlier the uncertainty in the probability $P_\nu$ of emitting $\nu$ neutrons in a fission event will result in a large uncertainty in the second and third moment of the neutron multiplicity distribution. For instance, a non-zero $P_6$ is common in spontaneous fission, which in the calculation of the third moment $P_6$ is multiplied by a factor 120 (see Table 2). Although $P_\nu$ values are correlated, the uncertainty in the third moment will be quite large, thus limiting the accuracy of the MC simulations.

**Table 2 Multiplication factors of each probability $P_\nu$ for the first, second and third moment of the neutron multiplicity distribution as a function of the neutron number $\nu$.**

| Moment | | |
|---|---|---|
| First | Second | Third |
| $\nu$ | $\nu(\nu-1)$ | $\nu(\nu-1)(\nu-2)$ |
| 0 | 0 | 0 |
| 1 | 0 | 0 |
| 2 | 2 | 0 |
| 3 | 6 | 6 |
| 4 | 12 | 24 |
| 5 | 20 | 60 |
| 6 | 30 | 120 |
| 7 | 42 | 210 |
| 8 | 56 | 336 |
| 9 | 72 | 504 |
| 10 | 90 | 720 |

## 4. Dead-time

The much higher count rates encountered in the measurement and consequently in the simulation of Pu bearing samples with respect to U bearing samples, result in an increased importance of dead-time related aspects. Already the first version of MCNP-PTA included the modelling of amplifier dead-time and that of or-chain mixing of the short pulses from TTL one-shot outputs. Subsequently, dead-time losses due to pulse pair resolution of the shift register electronics were added together with a complete modeling of the input de-randomising buffer. The latter results in time expansion of events with a higher multiplicity. This is particularly the case when a de-randomising mixer is used instead of the traditional or-chain. The de-randomising mixer (also modeled in PTA code) generates short pulse bursts (20 MHz), which are fed in the shift register electronics without any counting losses for the singles (totals). Some losses might be introduced for the doubles and triples rates due to the limited frequency of the shift register clock (4 MHz). Experimental verification of the correctness of the dead-time modeling of PTA is under way, while extension to the modelling of pulse shape and pulse height for proportional counters is being considered. This would enable to evaluate new counter designs with, for instance, a reduced dead-time.

## 5. Conclusions

The MCNP-PTA approach for the simulation of neutron multiplicity counters may be computing intensive (the use of a cluster is recommended [6]), but it allows the complete modelling (including dead-time) of a neutron counter, from source, via neutron transport and detection to the generation and analysis of electronic pulse trains.

Correct simulation of triples rates is a real challenge, it requires much more attention than it is the case for singles and doubles rates. When scanning pulse trains the direction in time is important. Moreover, the simulation results are much more sensitive to modeling inaccuracies (geometry, materials characteristics and nuclear data) due to non-linear dependencies.

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# Monte Carlo Techniques Applied to NDA: The JRC Ispra experience

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## Abstract:

*Since the 1970's, the Non Destructive Assay (NDA) group of the Joint Research Centre (JRC) at Ispra (Italy) has been in the forefront for developing nuclear radiation detection technologies and Monte Carlo methods in the field of Nuclear Safeguards. This paper reports on this experience in general and specifically on the application of the Monte Carlo methods of radiation transport and electronics simulation for the design, modelling and calibration of instruments as well as for the verification of nuclear materials. A number of applications and systems will be presented and reviewed.*

Keywords: Monte Carlo Techniques, Modelling, NDA, MCNP-PTA

## 1. Introduction:

Provided the Physics is correct, any radiation transport simulation is exactly analogous to the experimenter counting particles whereby each of many particles is followed from a source throughout its life to its death and the individual probabilistic events of a process are simulated sequentially. The probability distributions governing these events are randomly sampled using the transport cross-section data seeking to describe the total phenomenon and invoking the central limit theorem. With the physics getting better and better, coupled to the huge advances in computer capabilities and the ever richer cross section data at least for many elements of the periodic table (although there is a some reluctance around the word at providing adequate resources for more and better measurements), the Monte Carlo (MC) method in particle transport has become a very attractive (and often unavoidable) tool in our hands for many years now and in a large variety of fields (nuclear, health, industry etc..). Furthermore, measurements being not always possible or affordable, the MC techniques have become rather an absolute necessity in the same way as it was born in the Manhattan project (although its roots go further). This necessity is also recognized in our field of Nuclear Safeguards where, in addition to many geometry and practical restrictions and complications, suitable calibration standards are inexistent and access to nuclear installations and material to be verified are only too often unwarranted and too costly when at all possible.

However, a computer model or technique is only as good as its expert developer or user and the quality assurance applied. Hence, seeking to avoid any potential black box syndrome, assure and convince all communities involved (scientific, management and political), the model calculations must be validated, ideally using sound measurements and inter-comparisons (with measurements and other codes) whenever possible, coupled with a good sensitivity and error analysis.

In that respect, JRC-Ispra is privileged with a unique position in that it has available to it and often to its collaborators the Performance Laboratory (PLA) where a wide variety of nuclear material

samples (grams to kilograms of Uranium and Plutonium, radio-nuclide sources, etc..) which are used to test, calibrate instruments, validate and optimise Monte Carlo models.

Often overlooked in Non Destructive Assay (NDA) applications, however, are the uncertainties that result from an insufficient knowledge of the neutron yields and energy distribution spectra of Am-Li sources for instance in active interrogation applications [1] and the build up of $^{250}$Cf and $^{248}$Cm in old $^{252}$Cf sources which are widely used [2].

Over many decades, the Joint Research Centre (JRC) at Ispra (Italy) has been in the forefront for developing nuclear radiation detection technologies as well as Monte Carlo techniques and applying them in the field of nuclear safeguards in general and NDA in particular [3 - 7].

The 1970's saw the start of NDA at JRC Ispra with a computerised data analysis implementation of the shift register and the variable dead-time counter on a Laben computer and from 1984, W. Hage at Ispra and D.M. Cifarelli, performed the most comprehensive theoretical study in neutron correlation analysis [8-10]. In collaboration with Bondar of the Technical University of Aachen, Hage also built the first hardware multiplicity counter at JRC-Ispra, which was used for nuclear waste measurements in the late 80's.

It is however only in the 1990's that the Monte Carlo technique came into recognition and prominence within the field of NDA. The MCNP-PTA code [11], a Pulse Train Analysis (PTA) extension to MCNP [12], was developed at JRC-Ispra and has since been successfully applied to design, optimise, calibrate and cross-calibrate many NDA neutron counting systems, more recently to verify nuclear materials on site for the benefit of inspectorates and operators. The challenge of an online-verification using the MC method now seems a distinct possibility and a way forward.

This paper will give an overview of some such applications highlighting some of the points mentioned above, the main difficulties and restrictions encountered. Details for each system have been generally given in other publications to which the reader will be directed.

## 2. Monte Carlo and NDA

As in other fields, provided it is combined with sound measurements and model validations, the Monte Carlo technique has successfully become a powerful tool for the design and optimisation of NDA detector systems, their calibration and cross-calibration and the verification of nuclear material on site. The technique is often indispensable as standards for calibration may not be available, measurements are not possible, too expensive or too difficult, and access to sites is not always guaranteed. There is indeed now a real scope, although challenging, for applying the MC technique for on-line verification of nuclear materials.

For a single $^{3}$He tube, for instance, its response function can be written simply as:

$$R(E) = \sum_j \Phi_j A_s n^*_{He} V_{eff.} \sigma_{n,p}(E_j) \quad \text{where}$$

$\Phi_j$ = neutron fluence

$A_s$ = neutron source area

$n^*_{He}$ = $^3$He number density

$V_{eff.}$ = effective volume

$\sigma_{n,p}$ = $^3$He(n, p) cross - section

However, in NDA systems, the coincidence electronics (e.g. shift register) and multiplicity counting systems need to be modelled too in addition to simulating the neutron transport in the sample and detector system. MCNP-PTA code, a Pulse Train Analysis (PTA) extension to MCNP, was thus developed at JRC-Ispra for that purpose and recently MCNPX and other codes can also be used to simulate such counting systems. MCNP-PTA runs in two phases. Following MCNP run (phase 1), information such the originating event number, detector number where the neutron is detected, time elapsed from generation to detection is saved into a file for each neutron detected. The PTA part of the code generates (phase 2) the pulse train sequence which is analysed simulating the same logic as of the neutron analyser (e.g. JSR-12). Using the instrument settings (pre-delay and gate width) and including a realistic model of the dead time, the Totals, Accidentals and Reals rates are computed. MCNP-PTA has now matured and has been well validated and successfully applied to many systems.

## 3. Systems and Applications

### 3.1 MC Calibration of the FRM-II HEU fuel element.

The FRM-II reactor core, operated by the Technical University-Munich for research, medicine and technology, has one but unusual fuel element whereby the chain reaction is sustained by using highly enriched uranium to avoid neutron absorption by the non-fissile $^{238}$U. It uses a few kilograms of bomb-grade, highly enriched uranium (HEU) fuel and therefore nuclear safeguard and verification of the fuel element material are required. Neutron collars are installed for that purpose but there are no suitable standards which will allow their experimental calibration. The following procedure, based on computational calibration of the collar was adopted:

1. Models of detector and fuel element were developed using MCNP-PTA
2. HEU (MTR) reference materials were measured
3. The models were validated comparing measurements to calculations.
4. The response functions for the FRM-II fuel element were calculated
5. On-site verification of real FRM-II elements was performed at CERCA facility in France.

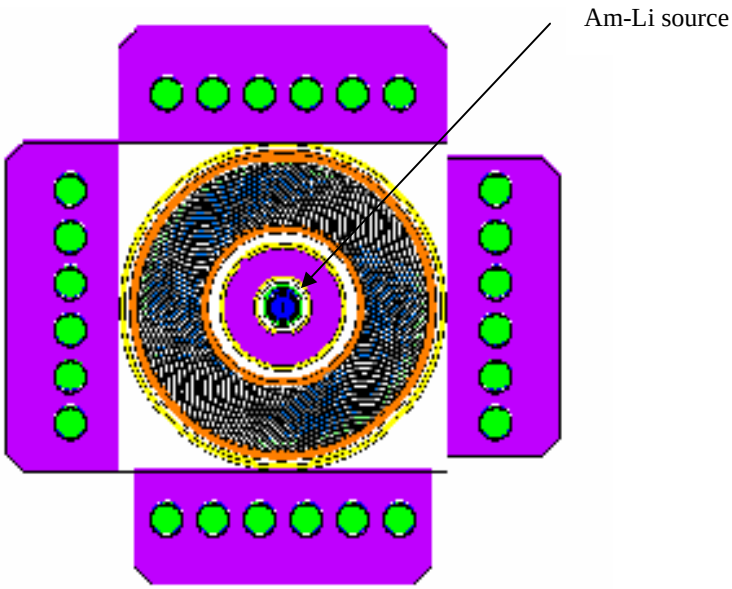


**Fig. 1: FRM-II neutron collar with 24 $^{3}$He tubes in the configuration whereby the Am-Li neutron interrogation source is placed at the centre.**

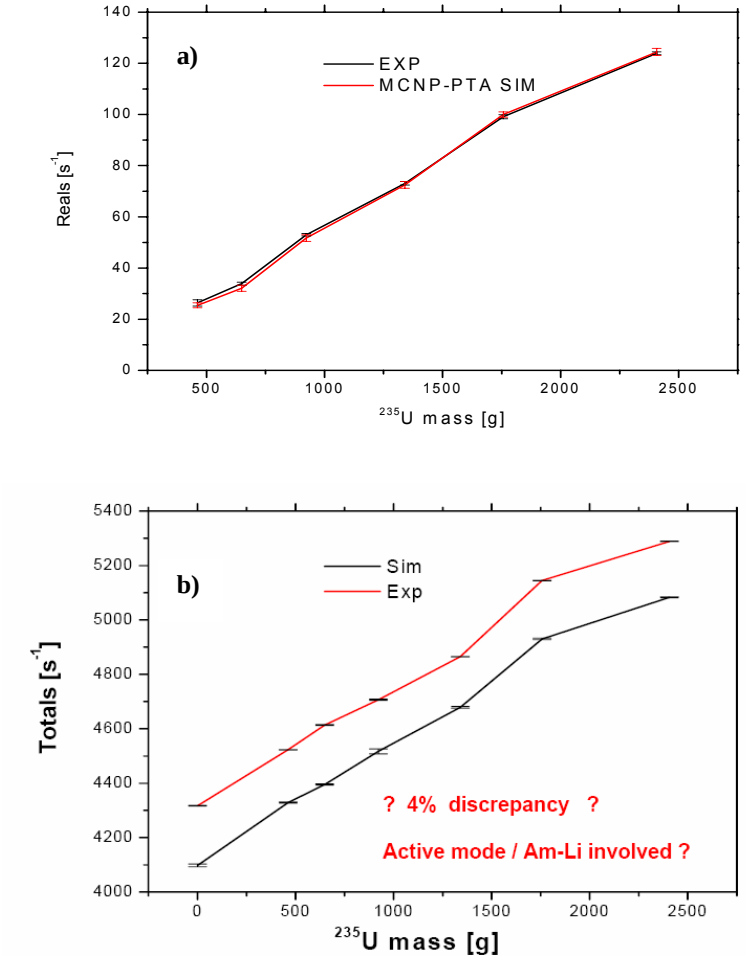


**Fig. 2: Measurements in PERLA (Laboratory JRC-Ispra) using 5 ESSOR fuel elements (93% enriched) inside the FRM-II collar – showing measured Reals rates (fig. 2a) and Totals (fig. 2b), compared to MCNP-PTA calculations.**

Two real fuel elements were measured at CERCA facility during PIV exercise and an agreement better than 2% was obtained between measurements and simulations for the Reals rates. However, it was found that the agreement was not so good (about 4%) for the totals rates and this is in line with findings elsewhere when Am-Li active neutron interrogation is involved. An investigation is underway (see H. Tagziria and M. Looman paper, this symposium) to improve knowledge of this source spectrum, which may be the cause of uncertainty especially at low energies. For the purpose of using the now validated model for the verification of the FRM-II fuel, a counting time of one hour will ensure a statistical uncertainty lower than 1%, which is very acceptable.

## 3.2 Extension of UNCL calibration

The application of Monte Carlo methods for the verification of LEU fresh fuel elements with enrichments greater than 4%, correction for Gd burnable poison rods and for model different, unusual or previously unknown fuel elements can be most valuable. The boundary conditions are often that one must preserve the cross-calibration concept and no modifications to the INCC software are allowed. In figure 3, we show the results of measurements carried out by the Los Alamos National Laboratory on reference PWR fuel elements using Uranium Neutron Collar (UNCL,

also labeled NCC) compared to MCNP-PTA calculations, thus extending the domain and range of the collars calibration, which is otherwise not possible without suitable (highly enriched) reference fuel elements. Cases where the MC technique comes to the rescue when the fuel elements to be verified are unusual and have no representative reference materials are well represented by the models of two WWER fuel elements shown in figure 4. Preliminary results of measurements carried out on WWER-440 fuel elements available at the Atominstitut in Vienna agreed within 2% with the MCNP-PTA calculations performed at JRC-Ispra.

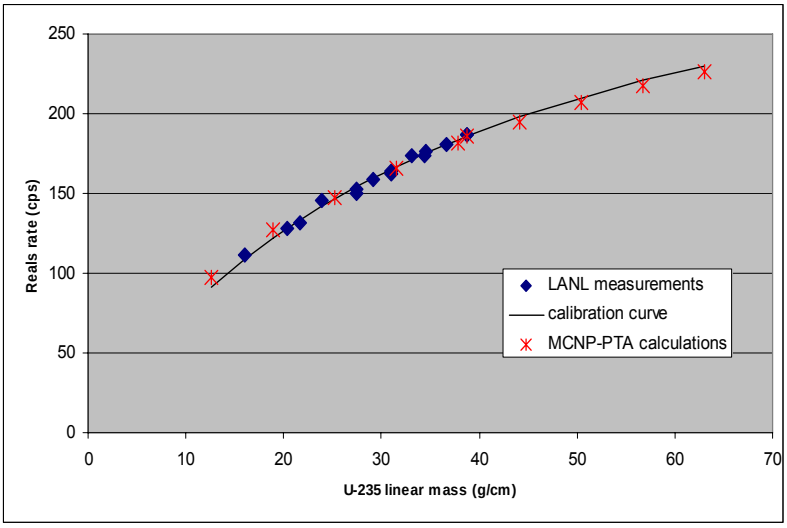


**Figure 3: MCNP-PTA calculations compared to the experimental curve obtained from LANL measurements with reference PWR fuel, extending the domain of calibration of UNCL.**

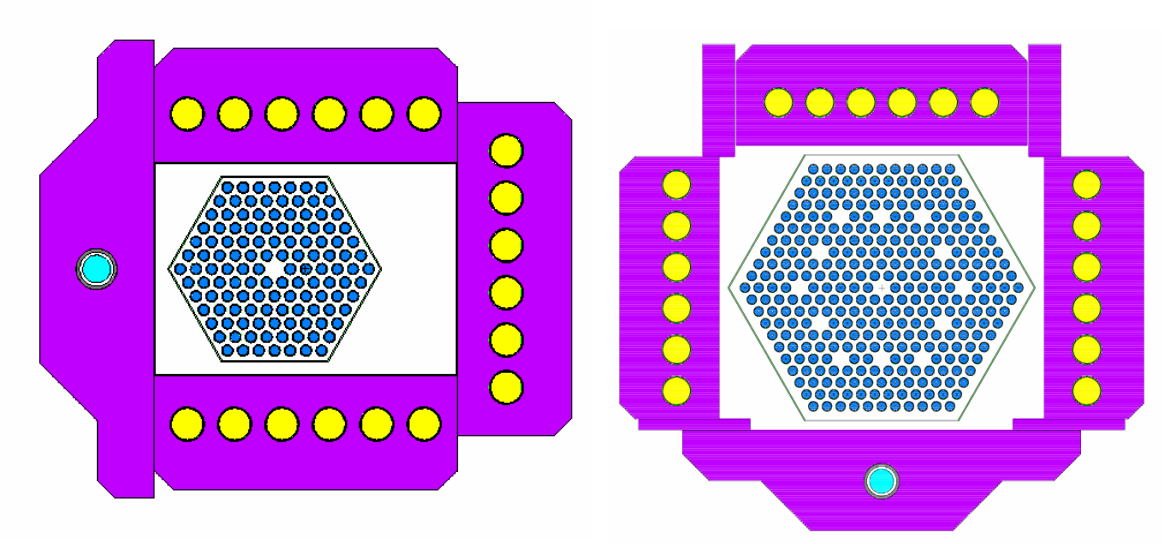

**Fig. 4: Model of a WWER-440, commercial BWR collar and one of WWER 1000 which is a commercial collar that has been modified to suit the different and larger fuel element.**

### 3.3 The D4-NCC counter at a MOX production Plant

A neutron coincidence collar (NCC), named D4, was installed for the verification of the flow of MOX fuel pins during production at a site where access is very restricted and there are no standards to be used for an empirical calibration of the counter [4]. Furthermore, the magazines may contain only intermediate rod configurations, i.e. partial loading, the cavity efficiency profile is strongly variable, the response functions do not only depend on the plutonium mass within the magazine but also on the geometry of the rod-loading pattern. It is neither possible to produce a classical calibration curve correlating mass to count rate nor feasible to pre-calculate every one of the large number of possible configurations. Thus a solution based on Monte Carlo simulation was adopted.

Both counter and associated electronics have been fully modelled using MCNP-PTA and the model has been validated using measurements carried out on site with Mixed Oxide (MOX) fuel pins and the efficiency of the counter for $^{252}$Cf radionuclide neutron source spectrum has also been calculated. Both types of measurements agreed to better than 1% with our MC calculations. A user interface has been developed to allow an inspector, on site, to define the loading pattern of the magazine, prepare the input file to MCNP-PTA and run the Monte Carlo simulation code for a given pre-declared loading pattern. One can thus verify reasonably quickly whether the measured and the calculated count rates, which correspond to a certain MOX mass, are consistent. The rationale behind using the Monte Carlo technique as a solution for the verification of MOX fuel elements has thus been thoroughly investigated in this project and shown to be sound, reliable and perhaps the best way forward in view of the limiting conditions on site, provided adequate model validations are performed.

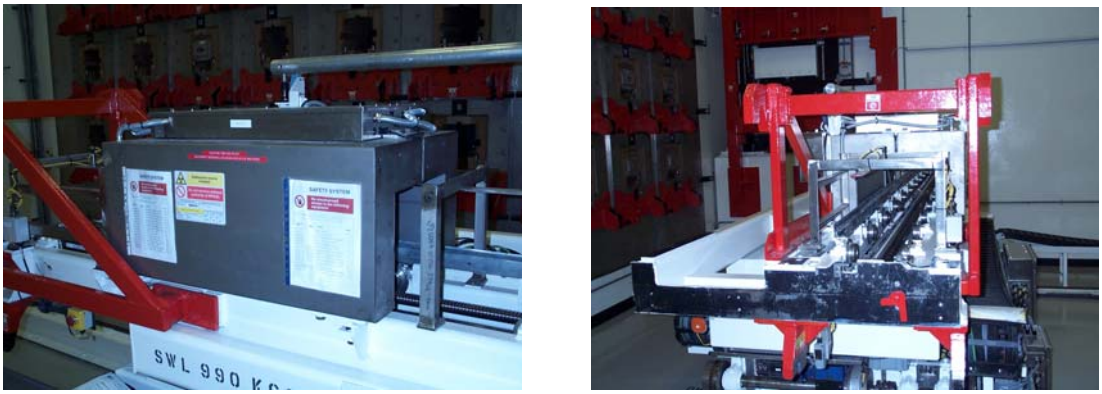


**Fig. 5: Photos of the D4-counter on site.**

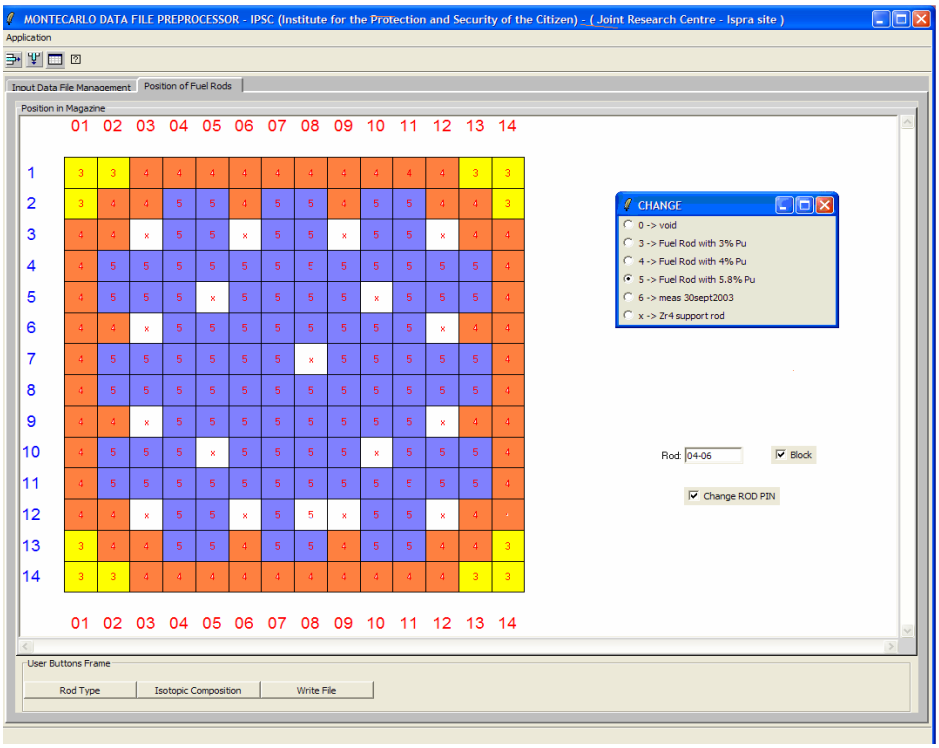


**Fig.6 : User interface page for fuel type selection in magazine.**

### 3.4 MC modelling and design optimisation of an NCC for verification of $PuO_2$ cans.

Another neutron coincidence collar intended for the verification of cans of $PuO_2$ powder at a MOX fuel fabrication plant has been designed and optimised using MCNP-PTA. The counter system, named D0, is of cylindrical shape containing 40 $^3$He cylindrical neutron detectors circularly disposed within polyethylene moderator through the middle of which a container of 4 or 5 cans of $PuO_2$ will pass for counting [5].

Since suitable calibration standards are unavailable, a procedure for numerical calibration of the detector was applied, using MCNP-PTA. From the operational point of view, the main difficulties derive from the rather severe constraints on physical size allowed the counter and from the fact that the cans of $PuO_2$ to be verified, will be contained within a container three times the detector effective length and will pass at a certain speed through the counter.

The response functions do not only depend on the plutonium oxide mass within the containers but also on the geometry of the oxide distribution, the density and filling patterns as multiplication effects for instance are not negligible. The counter's design has been established and optimised [5] and the system is currently in the production process after which it would be calibrated and the Monte Carlo model validated using a set of measurements carried out on site with a number of reference cans.

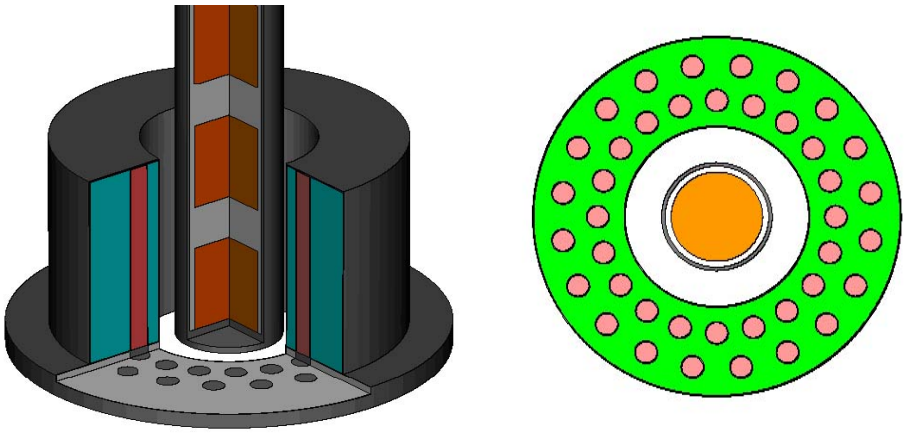

**Fig. 7: Model of the D0 detector in 3D and in the X-Y plan. The $PuO_2$ cans, 2/3 full in this figure, pass through the middle of the counter.**

## 3.5 The JRC Scrap Neutron Multiplicity Counter

As a part of a doctoral thesis a new Scrap Neutron Multiplicity Counter aimed for the verification inhomogeneous plutonium materials (e.g. scrap MOX) has recently been designed and its performance, optimised by Marin-Ferrer [7] using Monte Carlo modelling whereby the best materials that compose the counter, various geometrical parameters and many other relevant factors have been investigated and selected. Furthermore a new electronics, based on digitally processing the signals in order to reduce system dead times, has been utilised for the first time. The counter, which uses 124 $^{3}$He tubes to yield 63% $^{252}$Cf efficiency, has subsequently been produced on site and has been extensively tested and calibrated at the JRC-Ispra Performance Laboratory (PERLA) using a variety of well known Pu samples available to us and to our collaborators at ISPRA. Good agreements between measurements and the Monte Carlo models of the counter and its characteristics have been obtained. Details of the work are given in reference [7].

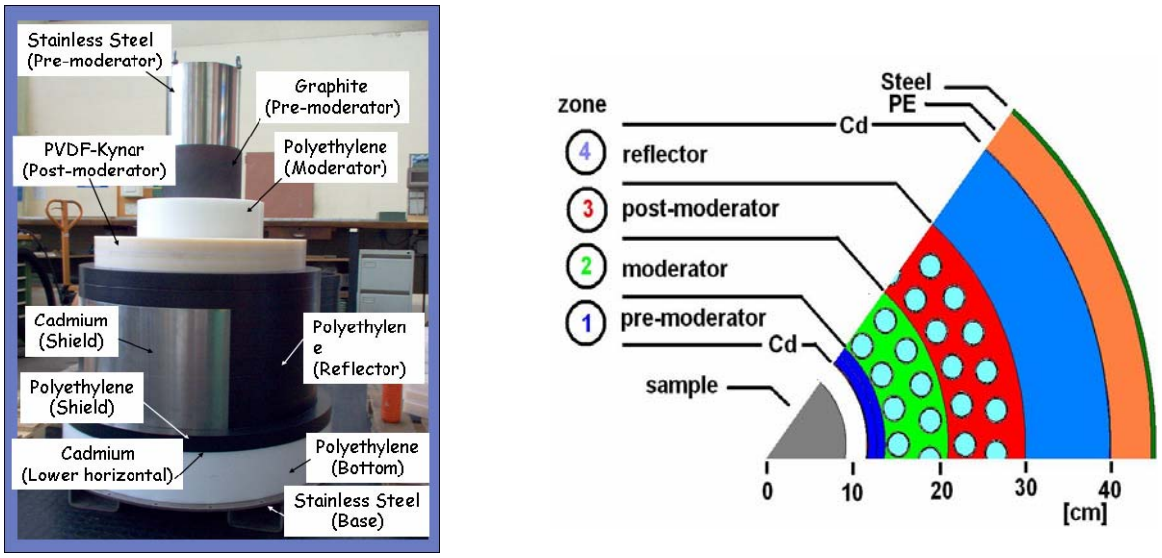


**Fig. 8: The JRC Scrap Neutron Multiplicity Counter [7] for the verification of Pu material.**

## 4. Conclusions

The Monte Carlo (MC) technique has become a powerful tool in nuclear safeguards and is often indispensable for the design and calibration of instruments, provided the calculations are combined with sound measurements, validations (with certified neutron sources and with high quality reference nuclear materials) and adequate Quality Assurance. There are situations where verification using MC technique may be one of the best ways forward if not the only one.

This is mainly due to the availability of richer cross section data which remains the basis for the physics involved, better source spectra (Am-Li, $^{252}$Cf although attention should be paid to any $^{250}$Cf and $^{248}$Cm buildup in old sources [2]), more powerful computers and better Monte Carlo techniques for radiation transport, with high quality physics such as the more accurate modeling of the fission process (multiplicity distributions).

At the Joint Research Centre in ISPRA (Italy) and elsewhere, the MC technique has been successfully applied for the design and optimisation of various NDA counting systems, their calibration and cross-calibration and the verification of nuclear material on site. The challenging scope now is to extend the technique to on-line verification of nuclear material. Results of our calculations agree generally well within 2% or better with measurements thus validating the techniques used and the models designed and applied.

The continuous validation of MC simulations with accurate measurements in our PERLA laboratory, where a wide variety of instruments and reference materials are available, has led to the discovery of important elements in the fields of application concerned and has helped to establish a broad knowledge base, two aspects which are fundamental in the building of high quality models of neutron coincidence/multiplicity counting and other applications.

The know-how and experience we have gained over the last decades has notably benefited our support programmes to Euratom and IAEA for instance, as well as the scientific community at large via various knowledge transfer schemes such training courses, collaborations, publications and on site technical support.

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# Analytical and numerical modelling of the detection statistics from a fissile sample

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***Abstract:***

*This paper investigates an analytical derivation of the distribution of the number of neutrons and photons emitted by a multiplying sample. The relationship between the statistics of the generated and detected neutrons and photons is also described. The analytical model described in this paper accounts for absorption and detection, thus extending and completing the model presented in previous studies. By using this new, improved model, one can investigate the relative feasibilities of measuring neutrons or gamma photons for the analysis of a specific fissile sample. In fact, whereas larger fissile masses will have larger multiplicities, resulting in an increase in both neutron and photon numbers, we show here that the self-shielding for gamma photons is much larger than that for neutrons. The results suggest that although photons have a larger initial multiplicity, neutrons might be more favourable to measure in the case of large samples because of the increasing self-shielding effect for the photons.*

**Keywords:** nuclear safeguards; master equations; number distribution; multiplicity; neutron and photon numbers.

## 1. Introduction

In non-destructive assay of nuclear materials, the study of the statistics of the number distribution of neutrons and gamma rays emitted by fissile samples is very important. The multiplicities of neutrons [1] and photons [2] generated in fissile samples with internal multiplication have been investigated in the past. Investigations of the effect of including absorption in the model have also recently been made. The starting point is implicit master equations for the generating functions of the neutron and photon number distribution. These have also been used in the past for calculating factorial moments. For practical reasons, factorial moments are usually only interesting up to third or fourth order. Using multiplicity and coincidence measurements, one can deduce the sample mass and isotopic composition of a given sample.

In contrast to the factorial moments, the probabilities $P(n)$ and $F(n)$ of emitting $n$ neutrons or gamma photons respectively, are interesting up to large values of $n$. The necessary number of terms of $p(n)$ that needs to be calculated is determined from the condition that the cumulative probability should be sufficiently close to unity. The number of terms can exceed 50 for both neutrons and photons, in some cases.

Including absorption in the model affects the statistics of the number distribution in several ways. For neutrons, the absorption will mean that some neutrons are eliminated from the fission chain and the number of observable (leaked) neutrons will be smaller. In the case of photons, the dependence is more involved, because both the absorption of neutrons and the photons themselves will affect the photon distribution. All of these effects will vary with the main parameter of the sample being investigated, which is the sample mass. An increase in mass affects the probability of both induced

fission, as well as absorption of both neutrons and photons for large samples in the few kg mass range.

The process of detection also affects the way one can observe the statistics of emission from a certain sample. There is always a given detector efficiency involved in the process of detection. Different ways of modelling this effect were investigated earlier [3]. The model in this latter publication assumes the detectors embedded into the sample. This model does not correspond to the physical situation; hence it leads to results that could be unrealistic. For instance a detector efficiency equal to unity means that there is only absorption in the sample and no fission, which is naturally incorrect. In this work we will present a different way of accounting for absorption and detection, which only concerns detection of the particles that are emitted by the sample. The results of the analytical calculations are compared to those from Monte Carlo simulations.

## 2. Theory

Master equations for the generating functions of the number distribution of neutrons and photons have been given in references [1,2]. In those models, as well as in the numerical simulations to which the model results were compared, the absorption of neutrons and gamma photons was not accounted for. It was assumed that each neutron has fixed probability *p* of inducing fission, or *1-p* of being emitted form the sample.

The probability generating functions (PGFs) *h(z)* and *H(z)* of $p_1(n)$ and *P(n)*, which describe the number distribution of neutrons generated by one initial neutron and one initial neutron event (spontaneous fission), respectively, are defined as

$$h(z) = \sum_n p_1(n)\, z^n \quad \text{and} \quad H(z) = \sum_n P(n)\, z^n, \tag{1}$$

respectively. The coupled backward master equations for neutrons read as follows [1]:

$$h(z) = (1 - p)z + p\; q_f\left[h(z)\right], \tag{2}$$

and

$$H(z) = q_s\left[h(z)\right]. \tag{3}$$

Here,

$$q_s(z) = \sum_n p_s(n) z^n \quad \text{and} \quad q_f(z) = \sum_n p_f(n) z^n \tag{4}$$

represent the generating functions of the number of neutrons generated in spontaneous and induced fission events, respectively. The number distributions are obtained by observing the fact that $p_1(n)$ and *P(n)* are the Taylor expansion coefficients of *h(z)* and *H(z),* respectively:

$$p_1(n) = \frac{1}{n!}\frac{d^n h(z)}{dz^n}\bigg|_{z=0} \quad \text{and} \quad P(n) = \frac{1}{n!}\frac{d^n H(z)}{dz^n}\bigg|_{z=0}. \tag{5}$$

The requirement to evaluate the expressions at *z=0* gives us some complications. If the derivatives were to be evaluated at *z=1*, we would have had factorial moments of the neutrons generated in spontaneous or induced fission; $\nu_s$, $\nu_f$. Now we arrive at modified moments instead that are defined by:

$$\frac{d^n\, q_\alpha(h)}{dh^n}\bigg|_{z\,=\,0} = q_\alpha^{(n)}(h)\big|_{z\,=\,0} = q_\alpha^{(n)}\left[p_1(n)\right] = \bar{\nu}_{\alpha n} \quad ; \qquad \alpha = s, f, \tag{6}$$

where $p_1(0)$ is the probability of having zero neutrons generated from one initial neutron. This probability needs to be determined since all other terms $p_1(n)$ and *P(n)* depend in a recursive manner

on that factor, and it occurs also in the modified moments. This means that the initial probability of having zero neutrons generated when starting with one initial neutron is vitally important and needs to be calculated to be able to find the other terms in the number distribution. Note can be made also to the fact that this probability will be highly dependent on the mass of the sample, and later, whether or not absorption and detection are included in the model.

The final expressions for the individual terms of *P(n)* show a formal equivalence to the factorial moments. By using the formulas for the number distribution we can simultaneously also obtain factorial moments to the same order as the value of *n*.

This same formal equivalence is found for the probability distribution and factorial moments for gamma photons. The probability distribution is given by [4]:

$$g(z) = \sum_n f_1(n) z^n \quad , \quad G(z) = \sum_n F(n) z^n, \tag{7}$$

where $f_1(n)$ and $F(n)$ describe the number of generated gammas starting with one initial neutron or one initial source event, respectively. The generating functions are defined implicitly from the following equations [2]:

$$g(z) = (1-p) + p\, r_f(z) q_f\left[g(z)\right] \tag{8}$$

and

$$G(z) = r_s(z) q_s\left[g(z)\right]. \tag{9}$$

Here $q_s(z)$ and $q_f(z)$ have been defined earlier but we also see the functions $r_s(n)$ and $r_f(n)$, which are the PGFs of the number distribution of photons from one spontaneous or induced fission, respectively:

$$r_f(z) = \sum_n f_f(n) z^n \quad , \quad r_s(z) = \sum_n f_s(n) z^n. \tag{10}$$

The nuclear quantities also give rise to modified moments for the photons, which are defined as follows:

$$\left.\frac{d^n r_\alpha(z)}{dz^n}\right|_{z=0} = n!\, f_\alpha(n) \equiv \overline{\mu}_{\alpha n} \quad ; \quad \alpha = s, f. \tag{11}$$

In the equations for the distribution $F(n)$ of the photons, the modified neutron moments that appear are different from the ones in the equations of the distribution $P(n)$ of the neutrons. This is due to the fact that the factor $p_1(0)$ in Eq. (6) will be exchanged for the factor $f_1(0)$ in the case of photons. The statistics of generated particles have been investigated earlier, and the master equations have also been used to find factorial moments. To be able to simulate actual detection statistics from fissile samples we need to include not only absorption into the model but also the detection process which takes place with a certain probability, often referred to as the detection efficiency, ε.

## 2.1. Neutron Distribution

In the neutron probability balance equation, the event of absorption can be included into the fission distribution, because it is formally the same as a fission event with zero neutrons generated. With this in mind, one can include the process of absorption by increasing the first collision probability, *p*, to a value *p'* representing the first collision probability, which hence accounts for both absorption and fission. At the same time, one has to appropriately modify (increase) the value of $p_f(0)$, and decrease the other $p_f(n)$ to maintain normalization. The formula to do this reads as follows:

$$p_f(n) = \frac{p' - p}{p'}\, \delta_{n,0} + \frac{p}{p'}\, p_f(n) \tag{12}$$

The first master equation will then read as:

$$h(z) = (1 - p')z + p'\, \tilde{q}_f\left[h(z)\right], \tag{13}$$

where $\tilde{q}_f(z)$ is the generating function of the $p_f(n)$ of Eq. (12). The factor *(1-p')* is the probability for a single neutron to leak out from the sample, as opposed to the earlier factor *(1-p),* which was the probability to not induce fission. Since comparisons will be made with Monte Carlo simulations using MCNP-PoliMi [5] we take the probabilities for inducing fission or being absorbed from that code, which contains the appropriate nuclear data tables.

In a similar way, the process of detection can be added by considering the neutrons that have been emitted by the sample, because it is only these that are available for detection. By using the above formalism, and introducing a detection probability ε, the process of detection can be accounted for by the use of the generating function $\xi(z)$ of the binary probability distribution of the number of neutrons detected per emitted neutron:

$$\xi(z) = \varepsilon\, z \;+\; (1-\varepsilon). \tag{14}$$

Here, ε is the detector efficiency for neutrons. The new generating functions that also include the detection process are given, from obvious considerations, by:

$$h_d(z) \;=\; h\left[\xi(z)\right] \quad , \quad H_d(z) \;=\; H\left[\xi(z)\right]. \tag{15}$$

The derivatives needed for finding factorial moments as well as the statistics change in a simple way

$$\frac{d^n h_d(z)}{dz^n} \;= \frac{d^n h(z)}{dz^n} \;\cdot\; \varepsilon^n \quad , \quad \frac{d^n H_d(z)}{dz^n} \;= \frac{d^n H(z)}{dz^n} \;\cdot\; \varepsilon^n. \tag{16}$$

For the factorial moments the full change is

$$\nu_{d,n} \;=\; \varepsilon^n \;\cdot \nu_n \tag{17}$$

The reason why these change in a very simple way is that they are evaluated at *z=1*. For the number distribution on the other hand, we have quantities that are evaluated at *z=0* which leads to the modified moments such as $\bar{\nu}_{sn}$, $\bar{\nu}_{fn}$. These modified moments depend on the $p_1(0)$ for neutrons and $f_1(0)$ for photons. However, as absorption and detection are incorporated, the probabilities of having zero particles emitted after starting with one neutron will change, and thus the modified moments change too, and so will the final probability distribution.

As before, the modified neutron moments are given by

$$\left.\frac{d^n\, q_\alpha(h)}{dh^n}\right|_{z=0} = q_\alpha^{(n)}(h)\Big|_{z=0} \;=\; q_\alpha^{(n)}\left[p_1(0)\right] = \bar{\nu}_{\alpha n} \qquad ; \qquad \alpha = s, f, \tag{18}$$

where the factors $p_f(n)$ are used. In the case of detection $p_d(0)$ will replace $p_1(0)$, which is solved from the *N*-th order polynomial

$$p_d(0) \;=\; (1 \,-\, p')\,(1 \,-\, \varepsilon) \;+\; p\; q_f\left[p_d(0)\right] \;=\; (1-p')\,(1-\varepsilon) \;+\; p\sum_{n=0}^{N} p_f(n)\left[p_d(0)\right]^n . \tag{19}$$

Using all these properties makes it possible to find the detection statistics from the Taylor expansions:

$$p_d(n) \;=\; \frac{1}{n!}\left.\frac{d^n h_d(z)}{dz^n}\right|_{z=0} \qquad \text{and} \qquad P_d(n) \;= \frac{1}{n!}\left.\frac{d^n H_d(z)}{dz^n}\right|_{z=0} . \tag{20}$$

The terms in the probability distributions can now be calculated recursively since the starting master equation is in implicit form. The analytic model now depends on the simple parameters; the probability to induce fission, which is a parameter that increases with mass, and can be calculated from the mass of the sample, provided the density is known; the absorption probability which is also depending on sample size and composition; and finally the detection efficiency, which can be changed to reflect

what type of detector is used, such as single fast scintillator detectors or large arrays of helium tubes in form of multiplicity counters.

## 2.2. Photon Distribution

To extend the model for photons to include absorption and detection requires more changes than for neutrons due to the fact that the generation of photons depend on neutrons. Therefore we need to consider the absorption of both the neutrons and photons. For the neutrons this was taken into account by changing the probability to induce fission, *p*, to *p'* as well as the probability distribution $p_f(n)$ and hence $\tilde{q}_f(z)$). However, for Eq. (8), the situation will be different. Since in this work we shall neglect capture gammas, leakage and absorption of a neutron will both lead to zero generated gamma photons, hence the parameter *p* in (8) remains that of the probability of inducing fission. The only notable change in the equations for photons when accounting for absorption of the neutrons will be that $q_f(z)$ needs to be substituted with $\tilde{q}_f(z)$, which treats absorption as an increased probability of generating zero neutrons in a fission.

Gamma absorption will be accounted for by the probability $p_L$ that describes the leakage probability for one single photon, likewise $(1-p_L)$ is the probability for a created photon to be absorbed and not escape the sample. The gamma capture will be accounted for by an additional master equation that describes the two mutually exclusive events of leaking or not leaking out from the sample for a single photon:

$$l(z) = l_\gamma\, z \;+\; (1 - l_\gamma). \tag{21}$$

Here $l(z)$ is the generating function of the binary probability distribution of gamma photons leaving the sample per initial photon. Due to the simple form of this relationship we can note that the factorial moments for leaked photons are simply the factorial moments for generated photons times a leakage factor:

$$\mu_{l,n} \;=\; l_\gamma^n \cdot \mu_n. \tag{22}$$

Further, the master equations for the leaked out photons are given as:

$$g_l(z) \;=\; g[l(z)] \quad , \quad G_l(z) \;= G[l(z)]. \tag{23}$$

The next step in the simulation of the statistics obtained from measurements is to incorporate the process of detection. Detection only concerns the leaked out particles, which thus can be taken into account in the master equations by adding an extra equation that describes the generating function of the probability to undergo detection or not:

$$\xi_\gamma(z) = \varepsilon_\gamma z \;+ (1 - \varepsilon_\gamma). \tag{24}$$

Using this equation we can obtain the detection statistics as:

$$g_d(z) \;=\; g\left[\, l\{\xi(z)\}\right] \quad , \quad G_d(z) \;=\; G\left[\, l\{\xi(z)\}\right]. \tag{25}$$

Since the probability distribution requires derivations to be evaluated at *z=0*, modified moments are created once again. They have earlier depended on $f_1(0)$ which is the probability of generating zero photons when starting with one initial neutron. Now these modified moments will depend on the new statistics of $f_d(0)$, i.e., the probability of having zero photons detected when starting with one initial neutron in the sample. The quantity $f_d(0)$ is calculated as the root of a finite degree polynomial:

$$f_d(0) = \; (1 - p) + p\; r_f\left[\, l\{\xi(0)\}\right]\; q_f\left[\, f_d(0)\right] \;= (1 - p) + p\left(\sum_{n=0}^{24} f_f(n)\left[\, l\{\xi(0)\}\right]^n\right) \cdot \sum_{n=0}^{8} p_f(n)\left[\, f_d(0)\right]^n. \tag{26}$$

In the case of gamma photons, when absorption and detection are included, the modified factorial moments are defined as follows:

$$\left.\frac{d^n r_\alpha\left[l\{\xi(z)\}\right]}{dz^n}\right|_{z=0} \equiv \bar{\mu}_{d,\alpha n}; \qquad \alpha = s, f, \tag{27}$$

$$\left.\frac{d^n q_\alpha(g_d)}{dg_d^n}\right|_{z=0} \equiv \bar{\nu}_{d,\alpha n}; \qquad \alpha = s, f. \tag{28}$$

These factors are straightforward to calculate but lead to expressions that contain sums, which have an increasing number of terms for higher order moments and are suitable for being calculated using computers and symbolic handling programs like Mathematica [6], which can handle the derivations easily.

## 3. Quantitative results

The number distributions were calculated for both neutrons and photons for plutonium metal spheres of varying mass. The composition of the spheres was 80wt% Pu-239 and 20wt% Pu-240, with a density of 15.9 g/cc. The values obtained with the analytical model were compared to simulations with the code MCNP-PoliMi [5].The nuclear physical constants such as the values of *p*, and the fission-parameters $p_s(n)$, $f_f(n)$, etc. were taken from MCNP-PoliMi runs and used in the analytical model. As expected, the probability to induce fission increases with the mass of the sample. The values of *p* can be found in the table below for the three samples for which we have performed calculations.

| Case | Mass (kg) | P |
|---|---|---|
| 1 | 0.335 | 0.06388 |
| 2 | 2.68 | 0.12464 |
| 3 | 9.047 | 0.18383 |

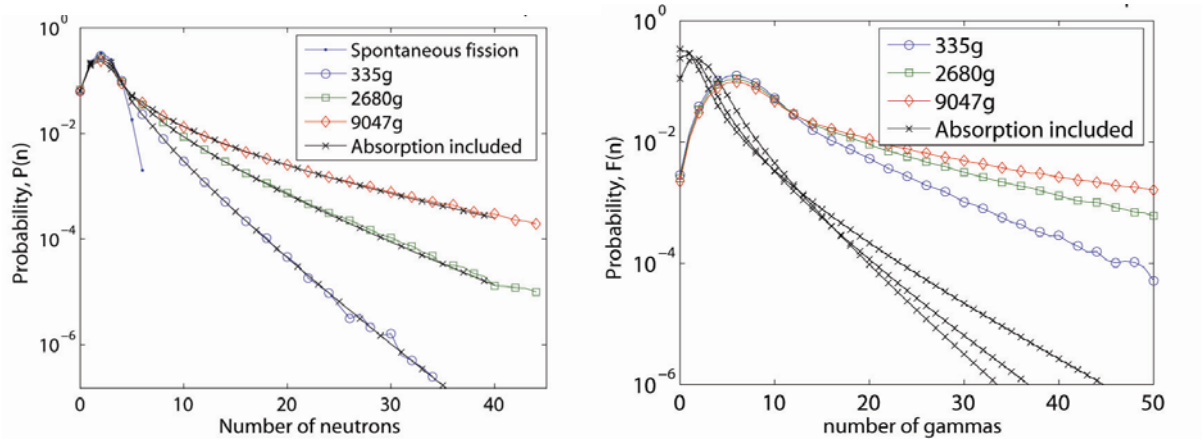


**Figure. 1** Comparison of the results from the analytical model and simulation results for the case with and without absorption. The comparison is made for three different sample masses.

Accounting for absorption in the model changes the statistics of neutrons very little while it has a large impact on the statistics of the photons. As seen in Fig. 1, the high-Z material does not stop many neutrons, but in the case of the photons the self-shielding effect is evident. The probability to have large bursts of particles decreases drastically in the case of gamma photons. Depending on the mass of the sample, the photons, which have a larger source multiplicity than the neutrons, can be emitted with lower probabilities compared to the neutrons.

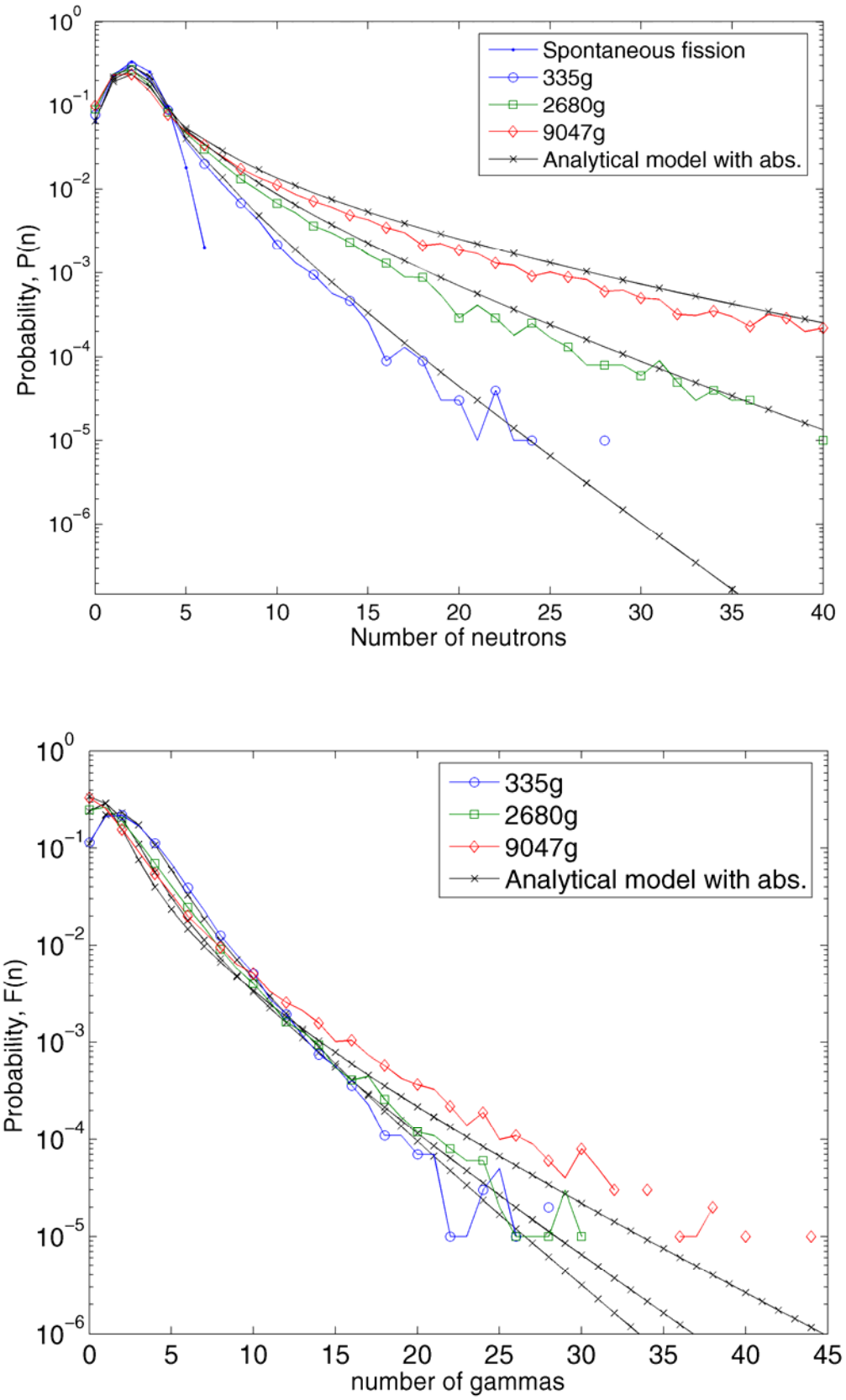


**Figure. 2** Comparison of the analytical model and Monte Carlo simulations. Results for both neutrons and photons are shown.

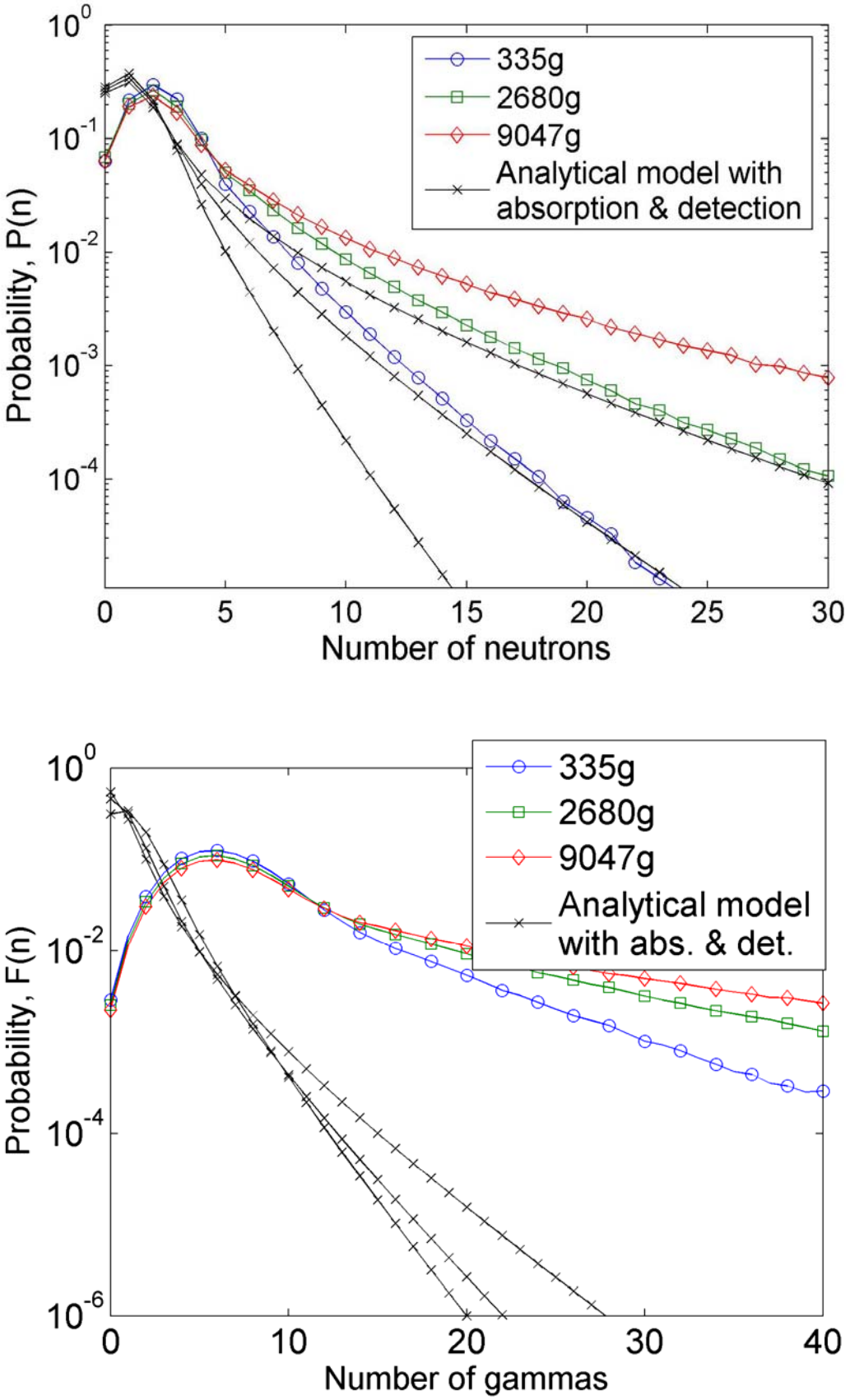


**Figure. 3** The number distribution of neutrons and photons when both absorption and detection are taken into account (detection efficiency = 50%). The data is compared to the data of

generated particles, i.e., without absorption and detection accounted for, calculated numerically using MCNP-PoliMi.

When observing the number distribution of detected neutrons and photons in the case of 50 % detection efficiency, Fig 3, we can see that the probabilities of detecting large numbers of particles decrease as expected, but we can also observe the fact that the probability of detecting singles and doubles is higher than the probability of generating singles and doubles that actually escape the sample, since the detected singles and doubles can also be the result of only detecting a fraction of the particles of a bigger burst.

In the case of gamma photons one can note the somewhat unexpected result that it is more likely to detect rather high multiplicities such as triplets and quadruplets for a lighter sample than for a heavy one. The reason is that, although the total amount of generated photons is higher for the heavier samples, the self-shielding effect counteracts this. The fast growing self-shielding (with increasing mass) constitutes of course a disadvantage. Note that these probabilities are per source event, and in a sample of higher mass the number of spontaneous fissions will be higher.

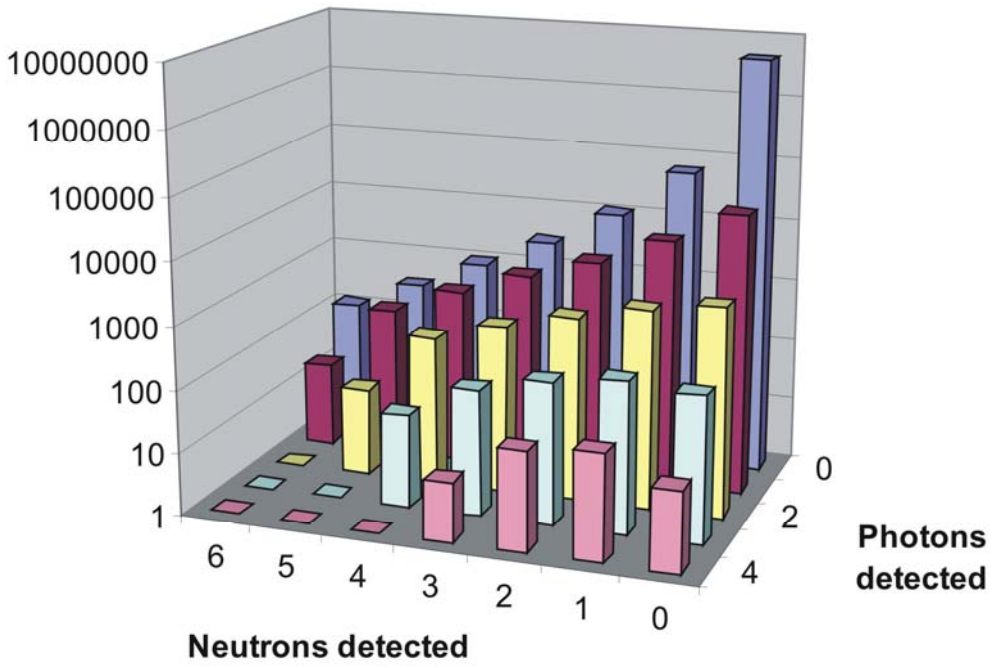


**Figure. 4** Monte Carlo simulation showing the number of occurrences for different combinations of particles detected. Six detectors were simulated with detection efficiencies equal to 0.00546 and 0.00549, for neutrons and photons, respectively.

It is expected that a complete understanding of the statistics as well as measuring of a large number of different multiplicities will allow for faster and more accurate analysis. The analytical models used here could also be used as indicators of what type of detectors are best practice to employ. In small samples, the larger photon multiplicities per fission event will lead to the emission of high photon multiplicities from the sample, and the detector efficiency will be the only limiting factor what regards measurement efficiency. For heavier samples, on the other hand, only neutrons escape the sample easily and neutron multiplicity counters might then be better to use. Finally the use of detectors capable of detecting both neutrons and photons provide more possibilities for analysis, as the joint probability distribution of both neutrons and photons can be used to obtain more information on the sample's characteristics.

## 4. Conclusions

We have developed an analytical model for the number distributions of neutrons and photons from a fissile sample. The model can be applied to predict the number of neutrons and photons that are generated, emitted, and detected. We obtained good results in comparisons with Monte Carlo simulations.
We used the symbolic computation code Mathematica to perform evaluations of the analytical model and to obtain higher order terms of the number distribution of the neutrons and photons. The results show that when absorption is accounted for, the number of photons emerging from the sample will decrease significantly, whereas the neutrons are not affected to the same extent. In the case of metallic samples, we showed that the multiplicities of photons emitted from the sample could decrease so much that neutron multiplicities become higher. With the introduction of the detection process into the model of the leaked neutrons and photons it is possible to simulate the detector response and find the probabilities for different multiplicities of both neutrons and photons. Using this information, one can assess different sample masses with regards to what type of emission has the higher multiplicity when using detectors for non-destructive assay of the material.

## 5. Acknowledgements

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# Session 16

# IS-Implementation II

# Evaluation of Integrated Safeguards Implementation

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## Abstract:

The Section for Effectiveness Evaluation (SEE) performs evaluation of IAEA safeguards implementation and produces the annual Safeguards Implementation Report (SIR) which is submitted to the Agency's Board of Governors. The SIR includes safeguards findings and conclusions for all States with safeguards agreements in force, based upon an evaluation of all the information available to the Agency in exercising its rights and fulfilling its safeguards obligations for that year.

This paper describes evaluation of integrated safeguards (IS) implementation, which is carried out for the purpose of the SIR.

A State-level integrated safeguards approach (SLA) is developed for a State with comprehensive safeguards agreement (CSA) and additional protocol (AP) in force. SLA takes into account State-specific features, such as the features of the State nuclear fuel cycle and the effectiveness of the State system of accountancy for and control of nuclear material (SSAC).

On the basis of this SLA and taking into account any recommendation for follow-up activities arising from the State evaluation process, the Agency develops an annual implementation plan (AIP) for the State concerned.

SEE performs evaluation of integrated safeguards implementation on the basis of SLA and AIP and prepares an annual IS evaluation report for each State under IS. This report includes an evaluation of the AIP, an evaluation of the extent to which the safeguards objectives have been achieved through the implementation of verification activities in a given year, an evaluation of the SSAC effectiveness and an evaluation of the verification effort spent.

**Keywords:** integrated safeguards; evaluation; State-level; effectiveness

## 1. Introduction

The aim of integrated safeguards is to provide the most efficient means to realize the full effectiveness of the strengthened safeguards measures.

Integrated safeguards are defined as: “the optimum combination of all safeguards measures available to the Agency under comprehensive safeguards agreement and additional protocols which achieves the maximum effectiveness and efficiency within available resources in exercising the Agency’s right and fulfilling its obligation in paragraph 2 of INFCIRC/153(Corrected)“.

To achieve this, the Agency implements the measures listed in both the comprehensive safeguards agreements (CSA) and the additional protocols (AP), providing credible assurance for both the non-diversion of declared nuclear material from peaceful nuclear activities and the absence of undeclared nuclear material and activities in the State as a whole.

Drawing a conclusion regarding the absence of undeclared nuclear material and activities is intrinsically more challenging than verifying the absence of diversion of declared material.

Integrated safeguards will not be implemented in a State until the initial conclusion of the absence of undeclared nuclear material and activities have been drawn, in addition to the conclusion in respect of non-diversion of declared nuclear material. The Agency will seek to re-affirm these conclusions annually both as an objective itself and as a condition for the continued implementation of integrated safeguards in that State.

For the implementation of integrated safeguards in a State - due to the enhanced assurance in the absence of undeclared nuclear material and activities for the State as a whole - the frequency and intensity of inspection activities at declared nuclear facilities and locations outside facilities (LOFs) may be at a lower level than those defined in the Safeguards Criteria. The Criteria are used to plan and to evaluate safeguards in States with no integrated safeguards implemented.

SEE State-level evaluation covers all verification activities performed by the Secretariat in the field and at Headquarters and is performed in order to determine the extent to which safeguards objectives, outlined below, have been achieved during safeguards implementation for each State in a given year. The evaluation is performed on an on-going basis throughout the calendar year. The results of the evaluation are reported in the SIR and provide a basis for the Board of Governors’ consideration of the safeguards conclusions and assessment of the effectiveness of safeguards implementation.

In order to facilitate the evaluation and comparison of the results for different States, a common set of three generic State-level safeguards objectives, applicable to each State, are defined as follows:

A. To detect undeclared nuclear material and activities in the State as a whole;
B. To detect undeclared production or processing of nuclear material at declared facilities; and,
C. To detect diversion of declared nuclear material.

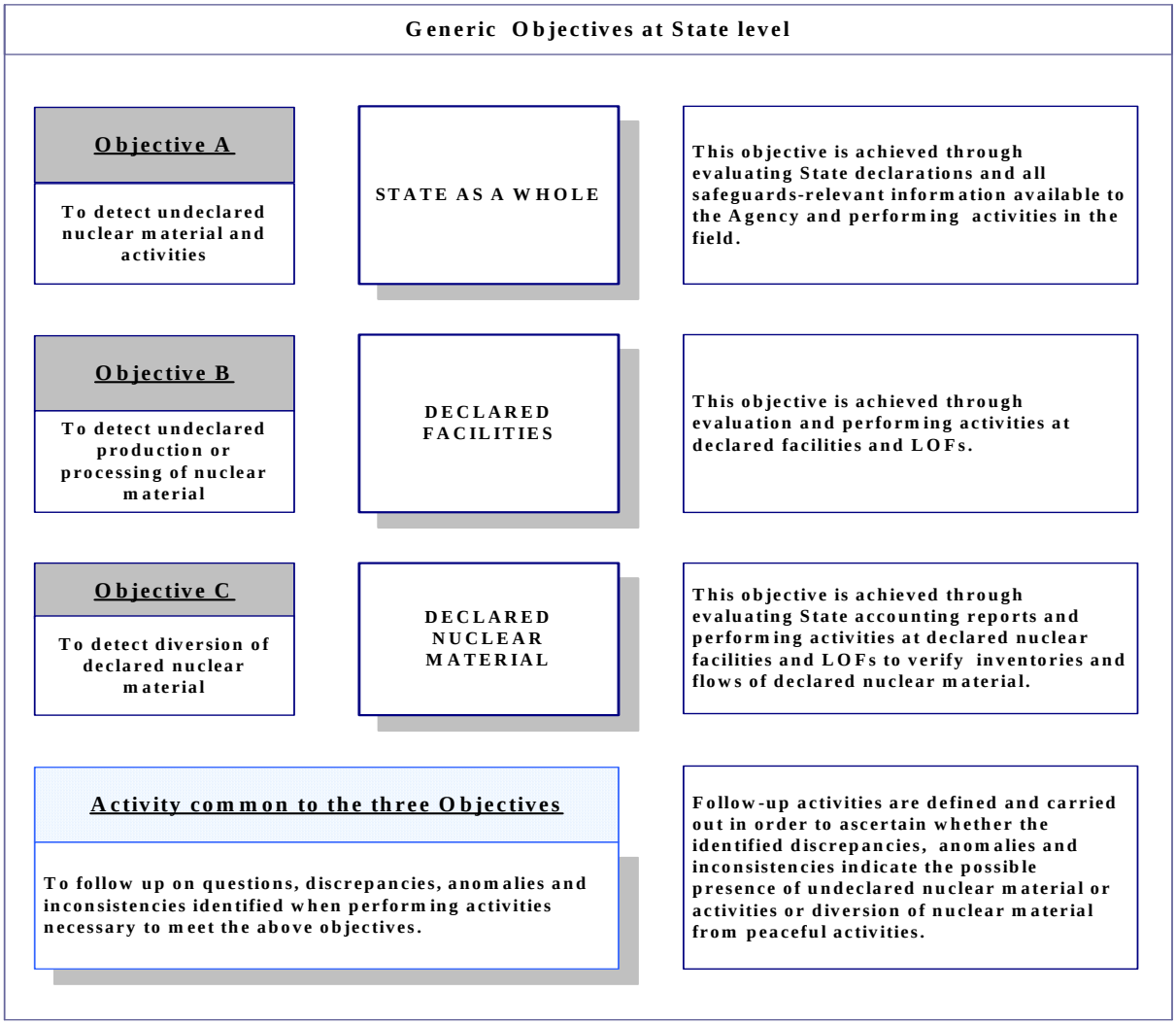
Each of the above objectives has also a common activity, i.e. to resolve anomalies and inconsistencies.

Although the generic State-level safeguards objectives would necessarily be the same for all States, it is important to define State-specific objectives based on an analysis of acquisition paths, i.e. the routes to acquire nuclear weapons usable material, State-specific nuclear fuel cycle features and characteristics and other information provided in the safeguards State evaluation report (SER) and to choose safeguards activities accordingly for a specific State.

## 2. State-level Concept for the Implementation and Evaluation of Safeguards

### 2.1 Generic State-level objectives and implementation of safeguards

The Agency plans and implements the verification activities in the field and at Headquarters to be able to meet the generic State-level objectives, as described in the figure below.




Although these objectives are interrelated, their separate analysis facilitates the planning and evaluation of safeguards implementation. The follow-up on questions, discrepancies, anomalies and inconsistencies when performing activities necessary to meet the objectives is an activity common to the three objectives.

- Objective A can be fully met only for a State with an INFCIRC/540 type Additional Protocol (AP) in force and implemented.
- Only for such a State is the Agency able to draw a broader conclusion that all nuclear material in the State has remained in peaceful activities.
- Objective B and C are achieved through evaluating all relevant information and, where applicable, implementing inspection activities at declared facilities and locations outside facilities (LOFs).

Where integrated safeguards are implemented, all the verification activities in the field or at Headquarters necessary to meet these objectives are defined in the State-level approach (SLA) and the annual implementation plan (AIP). Where no integrated safeguards are implemented, the relevant inspection activities are those defined in the Safeguards Criteria.

## 2.2 State-level Integrated Safeguards Approach

The SLA for a State sets out the safeguards activities to be conducted for a State, in accordance with the conceptual framework for integrated safeguards, comprising both in-field activities and work at Headquarters. It establishes the State-specific safeguards objectives which determine the level and focus of safeguards activities needed for the Agency to draw soundly based safeguards conclusions. It takes into account features and characteristics of the State's nuclear activities and capabilities identified in the State evaluation report (SER), the State-specific acquisition paths, the Agency's experience in the State, and the State-specific conditions for the implementation of safeguards measures.

The term 'conceptual framework' is used to describe the set of safeguards concepts, approaches, guidelines and criteria that govern the design, implementation and evaluation of integrated safeguards.

The State-level approach considers the State as a whole, and assesses wider aspects of a State's nuclear activities, such as:

- the structure of the nuclear fuel cycle, from uranium mines to nuclear waste repositories;
- the nature of fuel cycle-related research and development;
- the manufacture and export of sensitive nuclear-related equipment and material; and
- the effectiveness of the State's system of accounting for and control of nuclear material (SSAC).

The design of an effective and efficient SLA for a State should take into account the following elements considering the State-specific features and characteristics and adapting the model IS approaches for specific facility types:

- the basic principles established for IS;
- the three generic State-level safeguards objectives;
- the State-specific safeguards objectives;
- the State-specific conditions for optimizing the effectiveness and efficiency of safeguards measures, including the use of advanced safeguards technology, the use of unannounced or short notice inspections and the opportunities for cooperation with the State or regional system of accounting for and control of nuclear material (SSAC or RSAC) in implementing safeguards; and,
- the optimization of safeguards at facilities, including considerations of grouping of facilities that are related by location, type or function.

The SLA includes a plan for implementing complementary access (CA) at nuclear sites and other locations.

The main elements of a SLA for a State are: inspection, design information verification (DIV), complementary access, and information collection, review and evaluation.

At the end of 2006, State-level integrated safeguards approaches were approved for 14 States — Australia, Bangladesh, Bulgaria, Canada, Ghana, Hungary, Indonesia, Japan, Latvia, Norway, Peru, Poland, Slovenia and Uzbekistan — and integrated safeguards are being implemented for all these 14 States for the entire year 2007.

## 2.3 Annual Implementation Plan

The Agency develops an AIP for each State on the basis of the SLA (2005 was the first year that AIPs have been implemented). AIP includes a broader set of measures and requirements than those under traditional safeguards, such as:

- inspection activities;
- complementary access plans;
- design information verification (DIV) activities;
- information analysis at Headquarters; and
- list of questions, inconsistencies or anomalies for follow-up and resolution.

The safeguards activities planned for each year are to be specified in the AIP, which will reflect the State-specific safeguards objectives and associated safeguards measures, both in the field and at Headquarters.

The purpose of the AIP, in addition to identify the safeguards activities to be performed in a given year for a specific State, is to provide a means of converting the non-routine State-level integrated safeguards activities and SER recommendations into scheduled activities.

There are three parts to the AIP:

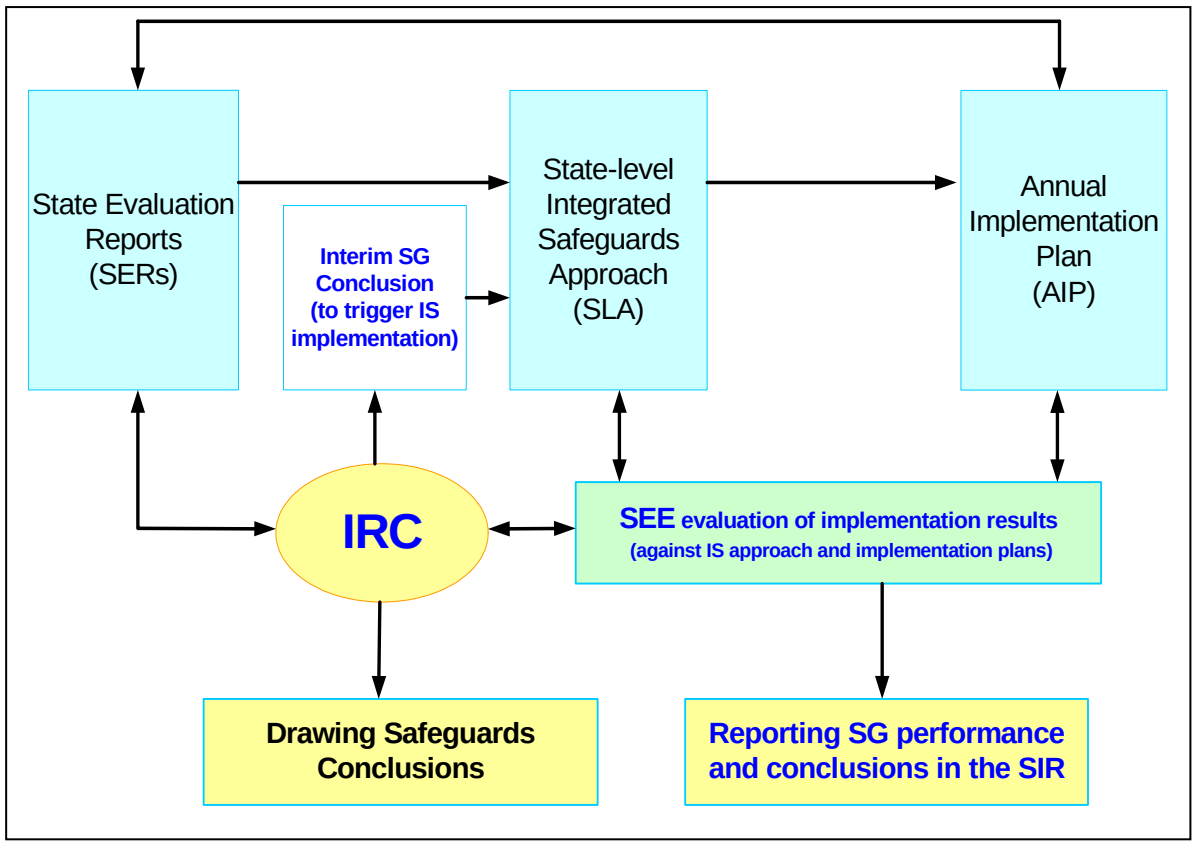
1. Nuclear material accountancy (NMA) verification and DIV: the facility- based safeguards activities.
2. Complementary Access: location, purpose and timeframe for CA activities.
3. Headquarters activities: specific activities outlined in the SLA or recommended in the SER (or/and by the interdepartmental Information Review Committee (IRC)) that are to be performed.

### 2.4 Inter-relationships between Safeguards Implementation Report, State Evaluation Report, State-level IS Approach and Annual Implementation Plan for the State Evaluation Process

The diagram below shows the interrelations between SIR, SER, SLA and AIP and the State-level evaluation process to enable the IRC in drawing conclusions about a State.

The Safeguards implementation report (SIR) is the IAEA Director General's annual report to the Board of Governors on the Safeguards Department's work, including its safeguards conclusions according to States' safeguards undertakings.

Under a State-level concept, the focus of the safeguards system has shifted from the facility-level to the evaluation of a wider range of information related to the nuclear related programme of each State as a whole. For most States, a Safeguards State Evaluation Report (SER) is produced each year. This report provides a snapshot of the Agency's knowledge and understanding in respect of the State.



### 2.5 Integrated Safeguards Evaluation Report

The report on the evaluation of the IS implementation is an important contribution to the overall process of drawing safeguards conclusions for States under IS.

This report prepared by the Section for Effectiveness Evaluation has four parts:

- a summary of the AIP evaluation with respect to the SLA for the year under review;
- an evaluation of integrated safeguards implementation and extent to which generic State-level and State-specific objectives have been met;
- an evaluation of the effectiveness of the SSAC; and
- the provision of information on the verification effort expended by the Agency in implementing integrated safeguards in the State.

While the IS evaluation report focuses on the implementation of the AIP which was agreed at the beginning of the evaluation period, it also evaluates effort undertaken as a result of changes to the AIP. Any changes to the approved AIP during the course of the year should continue to satisfy the overall safeguards objectives of the AIP and the SLA so that the changes do not become an issue during the year-end evaluation of the verification activities that are undertaken.

The SEE evaluation assesses both in-field activities and activities undertaken at Headquarters against the three generic State-level safeguards objectives, as applicable to individual States. The evaluation report summarizes the extent to which the verification activities implemented for a State have met the generic State-level objectives. It also indicates factors which may have an impact on the safeguards conclusion or affect the basis on which the safeguards conclusions are drawn.

## 3. Implementation and evaluation of Integrated Safeguards for 2006

In 2006, integrated safeguards were being implemented for the entire year for nine States (Australia, Bulgaria, Hungary, Indonesia, Japan, Norway, Peru, Slovenia and Uzbekistan). Safeguards implementation activities were carried out for these States in accordance with the SLA and the AIP approved for each individual State.

The activities carried out in 2006 included evaluation at Headquarters of all safeguards relevant information and verification activities carried out in the field:

(a) Headquarters activities relevant to objectives A, B and C:
- A State evaluation report (SER) for each of the nine States under IS was updated and reviewed by the IRC. Each SER includes an analysis of the State's declarations for internal consistency and for consistency with verification results and all other information available to the Agency. The SER also contains information on the status of the previously identified follow-up actions. When reviewing a SER, the IRC considers the significance of each finding and the extent to which it may affect the basis upon which the safeguards conclusions are drawn and makes recommendations for future follow-up actions accordingly.

(b) In-field activities relevant to Objective A:
- Complementary access (CA) was performed 43 times at sites and locations declared under Article 2 of the relevant additional protocols: 38 times with the objective " to detect undeclared nuclear material and activities", twice to confirm the status of declared decommissioned facilities, three times in order to resolve a question or inconsistency and once pursuant to article 8, i.e. with an access offered by the State.
- During CAs, 36 environmental samples (ES) and 3 destructive analysis samples (DA) were taken.

(c) Inspection activities relevant to Objective B:
- DIV was carried out at some facilities under construction or in the decommissioning phase.
- At research and power reactors capable of producing significant amounts of plutonium, containment/surveillance (C/S) and/or other unattended monitoring measures, complemented by unannounced or short-notice random inspection regimes, were implemented at all these facilities.
- At enrichment, reprocessing and associated conversion facilities, C/S and/or other unattended monitoring measures complemented by continuous, regular or random inspection regimes were implemented. Limited frequency unannounced access (LFUA) was performed 17 times at two enrichment plants.
- 33 ES were taken during inspections.

(d) Inspection activities relevant to Objective C:
- Evaluation of State accounting reports were carried out for 292 facilities and locations outside facilities (LOFs).
- PIVs were performed in 95 of the 97 facilities handling one significant quantity (SQ) or more of nuclear material. For the two facilities with no PIV scheduled, the State declarations were indirectly confirmed through implementing the short notice random inspection (SNRI) regime and taking into account the verification results at partner facilities.
- Material balance evaluations were carried out for all 97 facilities handling 1SQ or more of nuclear material; for 17 of them, handling material in bulk form, statistical analysis of material unaccounted (MUF), shipper-receiver differences (SRD) and their cumulative values had been performed.
- 16 facilities handling unirradiated direct-use material were subject to regular visit for timely detection purposes.

The Agency concluded that the evaluation and verification activities planned for 2006 for the nine States under integrated safeguards had been satisfactorily implemented. Nevertheless, some factors affecting safeguards performance were identified:
- The results of 45 ES taken during inspections and CAs in 2006 (or before) were still outstanding at the time of evaluation due to delays in sample analysis and/or evaluation.
- Some SLAs needed revision or further development in order to continue improving the effectiveness and cost-efficiency of safeguards implementation.

For 2006, the Agency estimated that the implementation of integrated safeguards resulted in saving of approximately 280 PDIs ( for 2005: 230 PDIs). This saving is mostly due to the abolition of scheduled quarterly interim inspections for irradiated fuel (timeliness detection period extended from three to 12 months) and to the random selection of interim inspections and PIVs for groups of facilities. The implementation of remote monitoring systems is also reducing the number of interim inspections at facilities with unirradiated direct-use material.

## 4. Integrated Safeguards conclusions

Having evaluating the IS implementation results, the Agency concluded that there was no indication of undeclared nuclear material and activities and no indication of diversion of declared nuclear material from peaceful use in the States. On this basis, the Agency concluded that, for these nine States under IS, all nuclear material remained in peaceful activities.

In addition, the Agency was able to draw this conclusion using less resources than would have been required if an integrated safeguards approach had not been implemented. The scope for savings in verification activities is greater for States with large, developed nuclear fuel cycle. Although IS have been implemented over the past six years gradually and in a limited number of States, they are being applied continuously and are functioning smoothly.

One of the Agency's goals in enhancing the effectiveness and efficiency of the safeguards system is to implement integrated safeguards in all States with significant nuclear activities and with comprehensive safeguards agreement and additional protocols in force. At the end of 2006, integrated safeguards were not implemented in 36 of these States. In addition, 19 States with declared facilities and/or LOFs containing nuclear material had comprehensive safeguards agreements in force but no additional protocols in force.

# Experience of Complementary Access in the European Union

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***Abstract:***

*Complementary Access is a new instrument provided by the Additional Protocol for the IAEA to fulfill its non-proliferation task. It has as an important role in the process of the Agency drawing conclusions about the absence of undeclared nuclear material and activities.*

*This tool is in use in the European Union since late 2004. A considerable number of Complementary Accesses have taken place, and the European Commission has participated in them, facilitating the Agency's activities in line with existing arrangements.*

*This paper reviews the experience of these verification activities in the Member States party to the Safeguards Agreement (INFCIRC/193). Practical arrangements for notifying and conducting Complementary Access, as well as problems encountered and lessons learnt, are discussed.*

**Keywords:** Additional Protocol, Complementary Access, European Union

## 1. Introduction

The protocol additional to the Safeguards Agreement (INFCIRC/193)[1][2] entered into force in the 13 non-nuclear-weapon States and the two nuclear-weapon States of the European Union on 30 April 2004. Beginning from that date the IAEA disposed of a new tool to carry out its task of verifying the completeness and correctness of the declarations provided pursuant to the reporting requirements and, more generally, the absence of undeclared nuclear material and activities. Complementary Access gives the Agency extensive access rights to a wide variety of locations which are not covered by inspections and design information verification visits under the Safeguards Agreement.

The protocol additional to INFCIRC/193 differs slightly from the model protocol INFCIRC/540, in order to take into account the role of the European Commission as the regional authority responsible for nuclear safeguards in the European Union. The Commission is responsible for the tasks involving nuclear material and activities, while each State carries the responsibility for tasks related to non-nuclear materials and activities. Responsibility is shared in matters concerning nuclear sites and waste. Annex III to the Additional Protocol permits the State to entrust to the Commission the implementation on its behalf. Ten of the thirteen non-nuclear-weapon States chose to use this so-called side-letter arrangement. The other three opted for the direct declaration to the IAEA of the parts that is considered as State responsibility.

Annex III also contains specific provisions for the Joint Research Centre (JRC), which is part of the European Commission and operates in four different Member States. Taking into account its specificity, in particular the independence from the host State authorities in planning and performing research, the Commission is fully responsible for the implementation in the JRC.

The two nuclear-weapon States of the European Union have each their Additional Protocol. They will remain out of this discussion because no experience of Complementary Access exists.

The ten States that joined the Union on 1 May 2004 did not immediately accede to the common Safeguards Agreement and its Additional Protocol. Their accession has been gradual, with Estonia, Slovakia, Slovenia and Poland covered by now, and the entry into force for Hungary and Malta scheduled for 1 July. For the others the procedures are also expected to be completed in near future.

The Additional Protocol introduced Complementary Access, a new verification tool for the IAEA, which significantly extends the Agency's access rights compared to the comprehensive safeguards agreements of the type INFCIRC/153. The provisions governing Complementary Access have now been in use for over two years and the Commission has accumulated a fair amount of experience of their implementation. In this paper we attempt to review this experience against the background of the legal provisions set out in Articles 4 through 10.

## 2. General Framework

### 2.1. Characteristics of Complementary Access

Complementary Access is an additional tool for the Agency to carry out its non-proliferation task of assuring that all nuclear materials are declared and used in declared activities. It differs from the 'traditional' safeguards inspections in many ways. Firstly, it is relatively new and we do not yet possess the wealth of experience that we have of safeguards inspections. Secondly, its form and conduct are less standardised and consequently less understandable to operators. CA is likely to take place more often in places and be conducted in ways that are – at least for now – less familiar to those involved. Its freer and more variable forms – access to places never accessed before, potential contact with a larger number of operators' staff, questions asked etc. – may cause confusion and, in the worst case, lead to unfounded suspicions and wasteful use of resources. Its implementation therefore poses new challenges to all parties involved.

Complementary Access is very much about communication between the Agency inspector and the operator's representative(s). As in all inter-personal communication, it is more likely that messages are misunderstood than understood in the way the sender intended them to be understood. The interpretation of any information given or statement made depends on the receiver's perspective and mental picture of the situation. There is bound to be large difference in the perspectives of the Agency inspector looking for proof of (no) undeclared material or activity and the operator's representative minding his production process or research activity. The inspector may be focussed on getting firm statements of the type 'such and such activity never took place' while the operator would either find it impossible to make such broad all-inclusive statements or tend to find even the questions intrusive: after all, what was done in the past was perfectly legal and did not require any declaration. It might simply be an overwhelming requirement to answer the question on the spot. Or the difficulty might be due to the all-embracing nature of the question.

We are sometimes confronted with questions that would put much smaller burden on the parties trying to provide response (operator, Member State and the Commission) if they were more specific and if more information was given about the indications on which the request for clarification is based. After all, an indication of an activity can only be an indication of concrete action being carried out by a concrete entity, private of public.

### 2.2. The Commission's Role

Under INFCIRC/193 the Commission is the 'State' counterpart to the IAEA for safeguards inspections and design information verifications. For the purposes of Complementary Access our role is different. Complementary Access is an IAEA activity where we act as facilitator. Our role is to observe, explain and be a link in the communication chain.

For the side-letter States the Commission is the focal point of communication concerning Complementary Access. It receives all notices and transmits the information to the operator and the National Authority responsible for the Additional Protocol. Euratom inspectors accompany the IAEA during the Complementary Access. Communications after the visit, including possible questions or requests for further clarification, arrive to the Commission, which prepares answers in cooperation with

the operator and/or the State. With regard to the non-side letter States, the Commission receives advance notices and is involved in the areas under its own or shared responsibility.

The experience that we have accumulated over the past couple of years permits us to contribute by our presence to the equal and just implementation of the Additional Protocol throughout the countries of the European Union. Where needed, we can intervene and explain to our Member State actors their obligations and their rights in this context and we can facilitate the work of the IAEA in the EU.

### 2.3. Some Descriptive Statistics

The IAEA made the first Complementary Access on 21 December 2004. The verification activities gained momentum from the second half of March 2005, and by the end of that year 45 had been performed in the 13 NNWS. This corresponds to about a fourth of the total number of Complementary Accesses conducted in the whole world in 2005. With 25 additional Complementary Accesses in 2006, the Agency's verification activities on the ground slowed down clearly in numbers. The figure for 2006 includes 5 Accesses in three of the new Member States of the 2004 enlargement.

By the time of writing 74 Complementary Accesses had taken place. As regards the split between the side-letter States and non-side-letter States, the former account for 56, leaving 18 to the latter. A Euratom inspector was present in each. A representative of the national safeguards authority attended almost all Accesses in the non-side-letter States and some, although fewer, in the side-letter States. Chart 1 illustrates the relative number of Complementary Accesses made in the Member States of the European Union in relation to the number of sites in the same Member States.

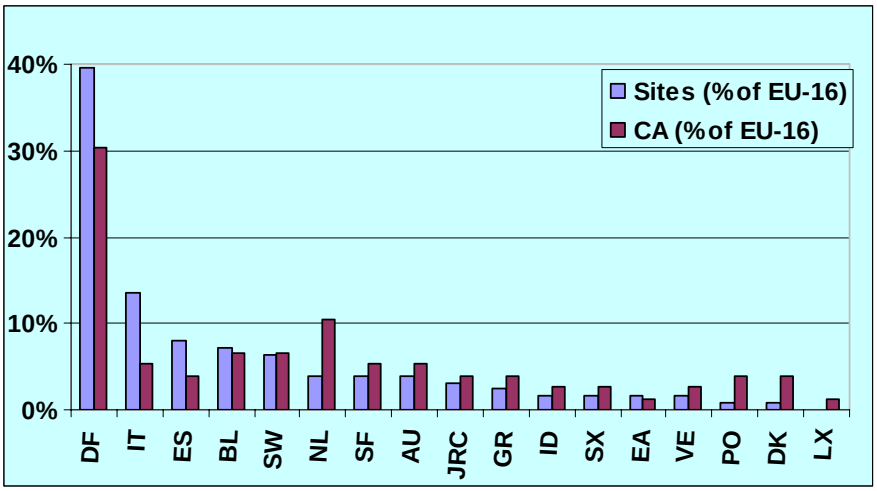


**Chart 1:** Complementary Access by Member State.

## 3. The Legal Provisions and How the IAEA Has Used Them

Articles 4 through 10 open to the Agency many new doors for access, in addition to what is provided for in the Safeguards Agreement. At the same time, these Articles lay down the conditions and modalities governing the use of the enhanced verification powers. Our discussion of the experience gained in the implementation of Complementary Access in its legal framework will start with some general and procedural matters. Then we will turn to reviewing on which bases the Agency has used its powers. It is interesting to see which among the available provisions have actually been used in access requests. Finally activities undertaken and reporting provided will be reviewed.

### 3.1. General Principle and the Purpose of Access

Article 4.a. sets out the general principle for the implementation of the provisions concerning Complementary Access by stating that the Agency *shall not mechanistically or systematically seek to verify the information* the Member States and the Community are required to provide to it. The

requirement that verification should not be mechanistic or systematic can be seen from two sides. From the one side, it asks the Agency not to use resources to try to verify all that has been declared to it under the Additional Protocol. From the other side, it is a call to make use of unpredictability, which would be lost if reports were verified systematically.

Much ink and paper could be consumed to discuss what is and what is not meant by mechanistic and systematic verification. With an agreed definition, statistical techniques could be used to test whether Complementary Accesses in the past show mechanistic and systematic patterns, but this would be possible only within the IAEA, as it uses a variety of information sources in its work. Some might see a sign of mechanistic or systematic implementation if they observe at least one Complementary Access in (almost) every Member State every year, or if three or four in a given type of facility in different countries took place over a year. However, there would surely be no-one to ask the Agency to behave in the opposite way, by entirely randomising its activity.

## 3.2. Advance Notice

### 3.2.1. Notice Period and Notification Procedure

Article 4.b. provides for the minimum period of advance notice to be given by the Agency. This is 24 hours except when access is sought in conjunction with a safeguards inspection or a design information verification visit.

So far in the European Union the notice was given 24 hours before the start in 48 cases of the total 74 accesses made. In some of these – less than half a dozen – the access was effectively made in conjunction to a safeguards inspection. In 26 cases the Agency inspector handed over an advance notice during a safeguards inspection or design information verification, to obtain access within 2 hours or more.

The procedure for notifying Complementary Access depends on the notice period. In the case of 2-hour notice during an inspection, the IAEA inspector hands the advance notice to the Euratom inspector, who checks its validity and passes it to the person responsible for the Additional Protocol on site. Often this person is the Site Representative designated in line with the Euratom Regulation [3] as person responsible for contacts between the European Commission in Luxembourg and the operator(s) of the site. The notice handed over to the Euratom inspector is currently the only advance notice provided by the IAEA where request for Complementary Access is made in conjunction with an inspection. In most of these cases, the request for access has been made for and the activities started within the minimum notice period. In few cases the notice was given earlier, perhaps better accommodating the Complementary Access to the inspection activities.

Where access is requested with a notice period of at least 24 hours, the procedure depends on the location to be accessed. In practice all Complementary Accesses so far have been made to locations involving nuclear material or activities. In such cases the advance notice arrives by fax to Luxembourg and is then forwarded to the concerned Site Representative and the State Authority, with a cover page informing about the name of the Euratom inspector who will accompany the Agency during the Access. In order to ensure that the information is passed quickly and the formal notice received by a person who can make the practical arrangements for access, we always try to reach by phone the Site Representative or, in his absence, another person knowledgeable of the Additional Protocol and Complementary Access immediately after the advance notice has arrived.

In line with the provision of Article 4.e. all Complementary Accesses have been performed during the regular working hours of the location subject to access. While there is no formal requirement to give advance notice during normal working days, this has been the practice in all except one case. We submit that apart from exceptional urgency, giving notice on a public holiday or otherwise outside normal working hours would go against the spirit of the Additional Protocol and place an unnecessary burden on operators, Member States and the Commission.

### 3.2.2. Notice Content

Article 4.c. requires that the advance notice be in writing and *specify the reasons for access and the activities to be carried out*.

What are the *reasons for access* that should be specified in the notice? The Agency has interpreted *reasons for* as synonymous to the purpose of access, which is expressed in Article 4.a.(i)-(iii) using the words *in order to*: *…the Agency shall have access to […]Any location […]* ***in order to*** *assure the absence of ….*

The requirement of giving reasons for access is addressed in the advance notice by repeating the text of Article 4.a. that follows the phrase *in order to*. Although the Agency's preference for a very generic statement of 'reasons' is understandable on some grounds, one must acknowledge that no information specific to each request is given. Therefore, if the intention of Article 4.c. is that the Agency provide information, we must conclude that this is not happening.

To make Complementary Access successful, more detailed information concerning the reasons or the objective is needed. When and how much information the Agency will give has its importance. Our inspector will ask for the objectives in the meeting held immediately before the start. Sometimes it might be useful for the Agency to provide beforehand a more precise indication of their focus, in this way giving a possibility for the operator to prepare and thereby increasing the chances of getting the expected clarifications. This is particularly relevant in places where the Agency intends to raise questions about the past or otherwise seeks very specific clarifications where the presence of an expert would be needed. Little experience of such advance information exists, although questions of this type have been raised in a number of cases. Indeed, if there is no specific reason to limit the notice period to the minimum, the additional leeway in arranging for activities and preparing information sources would seem to well justify giving notice earlier than 24 hours before access.

Interestingly Article 4 does not explicitly ask the Agency to specify the location where the access is to take place. While it is true that naming the location is unavoidable and the requirement to do so could have been left out as superfluous, this omission may be the reason for a curious feature in the advance notice. The location to be accessed is mentioned in an indirect way, without reference to Article 5, which lays down the permissible locations. Instead, the advance notice refers to Article 4.b.(i) or 4.b.(ii) as the basis for access although these only concern the notice period.

As regards activities to be carried out, the advance notice lists all permissible activities. This ambiguity is understandable, as it gives an additional degree of freedom to the Agency without usually creating additional burden to the operator. The activities do not in the majority of installations require specific preparation. If advance preparation is required, which could the case for environmental sampling in a hot cell for instance, we expect the IAEA to provide the advance notice in a timely manner.

### 3.3. Bases for Access Request

The locations where the Agency has the right to make Complementary Access are set out in Articles 5 and 8. Article 5 deals with requests where the Agency takes the initiative, discussed in sections 3.3.1-3.3.3, while section 3.3.4. relates to Article 8, which provides for a voluntary offer from the Member State.

With regard to Article 5, the locations open for Complementary Access can be divided into three broad categories. First, those where nuclear material is used or was used in the past. These are covered in Article 5.a. and will be referred to as 'nuclear locations'. Access requests to locations only declared under the Additional Protocol provisions not relating to nuclear material are provided for in Article 5.b. These refer to R&D that does not involve nuclear material, the manufacturing of materials and equipment referred to in Annex I, as well as the imports of materials and equipment referred to in Annex II. These will be discussed under the heading of 'non-nuclear locations'. Finally, the Agency can request access to any other location on the basis of Article 5.c. Such locations include those that have not been declared by the Member State or the Community. They will be called 'Other locations'. Table 1 summarises the locations and the purpose of Complementary Access in each.

| Location | Purpose of Access (Article 4.a.) |
|---|---|
| *5.a.(i) Any place on a site on a selective basis.* | *to assure the absence of undeclared nuclear material and activities* |
| *5.a.(ii) Any location identified under Article 2.a.(v)-(viii) on a selective basis.* | *to assure the absence of undeclared nuclear material and activities* |
| *5.a.(iii) Any decommissioned facility or decommissioned location outside facilities where nuclear material was customarily used to the extent necessary for the Agency.* | *to confirm, for safeguards purposes, the Community's, or, as appropriate, a State's declaration of the decommissioned status of a facility or location outside facilities where nuclear material was customarily used* |
| *5.b. Any location identified by the State concerned under Article 2.a.(i), Article 2.a.(iv), Article 2.a.(ix)(b) or Article 2.b., other than those referred to in paragraph a.(i) above.* | *to resolve a question relating to the correctness and completeness of the information provided pursuant to Article 2 or to resolve an inconsistency relating to that information* |
| *5.c. Any location specified by the Agency, other than locations referred to in paragraphs a. and b. above.* | *to resolve a question relating to the correctness and completeness of the information provided pursuant to Article 2 or to resolve an inconsistency relating to that information* |

**Table 1:** Locations and the Purpose of Complementary Access.

### 3.3.1. Nuclear Locations

The locations currently or formerly in possession of nuclear material are divided further into three groups. The first group – sites – consists of facilities and locations outside facilities, most of them subject to safeguards inspections. 2-hour notice is possible only in these. Almost all Complementary Accesses (69 of 74) have been conducted on sites. Some sites, however, only contain installations where regular safeguards inspections do not take place. Sites were accessed in 43 cases with a 24-hour notice. In a small number of these cases the access was made in conjunction to an inspection.

The large number of accesses to locations subject to regular safeguards inspections might raise the question whether Complementary Access was really used as a tool 'complementary' to the verifications made in nuclear installations under the Safeguards Agreement. If we use as a rough measure of 'complementarity' the type of places within the sites accessed during Complementary Access and compare them to the places accessed during inspections, we see that Complementary Access has indeed been used as a complementary tool. The places accessed have been generally different, and Complementary Access has mostly taken place in buildings without safeguarded material or activities. In 5 cases, however, the Complementary Access activities were those typically performed during a design information verification.

The second group of nuclear locations consists of locations with nuclear material but out of safeguards inspection activity because of an exemption or because they are at the front- or back-end of the nuclear fuel cycle. Only 3 Complementary Accesses have been made to these categories of locations, all to locations with mines, either currently operating or permanently closed-down.

The third group of nuclear locations covers locations with a facility or a LOF that has been declared to the Agency as decommissioned. The purpose of access is to confirm that the declared status corresponds to the reality. No Complementary Access has so far taken place on the basis of this Article. The practise adopted in the Euratom area has been not only to declare installations as decommissioned but to obtain the Agency's confirmation of the status after a common verification. The common verification activity was started well before the entry into force of the Additional Protocol, the intention being not to create sites where installations were mature for being confirmed as decommissioned. This practice gives legal certainty to both the former and the possible new operators.

A question that has been raised before but perhaps still merits to be asked is whether the text of Article 5.a.(iii) should be read as granting the Agency the right to continue to make status verifications even after having concluded that an installation is decommissioned. What would be the value of the initial conclusion regarding the decommissioned status, if the Agency would be expected at any moment of its choice make another similar verification? The question is valid in particular for locations where the premises have been taken over by entities whose activities are unrelated to the nuclear sector and which are therefore not known to the national or regional safeguards authority. Should a separate list of such entities with contact information be held up-to-date for the eventuality that the Agency might request access by giving notice only 24 hours in advance? Does not the essence of confirming the status of an installation as 'decommissioned' entail that, at the time of verification, all equipment essential for the nuclear activity had been definitely removed? Would it not be reasonable to take the step to conclude that the location has thus definitely ceased to exist and cannot be resuscitated as far as nuclear safeguards are concerned? If the same or another entity were to start a nuclear fuel cycle related activity subject to safeguards in the same geographical location, this would be handled as a completely new operation. If the Agency suspected undeclared activity in a location that had been verified as decommissioned, it could raise a question and obtain access under Article 5.c. (see section 3.4.3.)

Chart 2 gives an approximate idea of the types of installations which have been accessed under the provisions for Complementary Access. The classification is somewhat arbitrary as it is at the level of an installation. Keeping in mind the limitations of the approach used, we can get a rough picture of the types of locations accessed so far. Not surprisingly, installations with research related activities have received the largest number of Complementary Accesses. For a more accurate picture, we would need to descend to the level of buildings accessed. At that level, the share of 'waste related' would grow and a fair number of cases would move from 'enrichment' to 'research-related' or ‘manufacturing’.

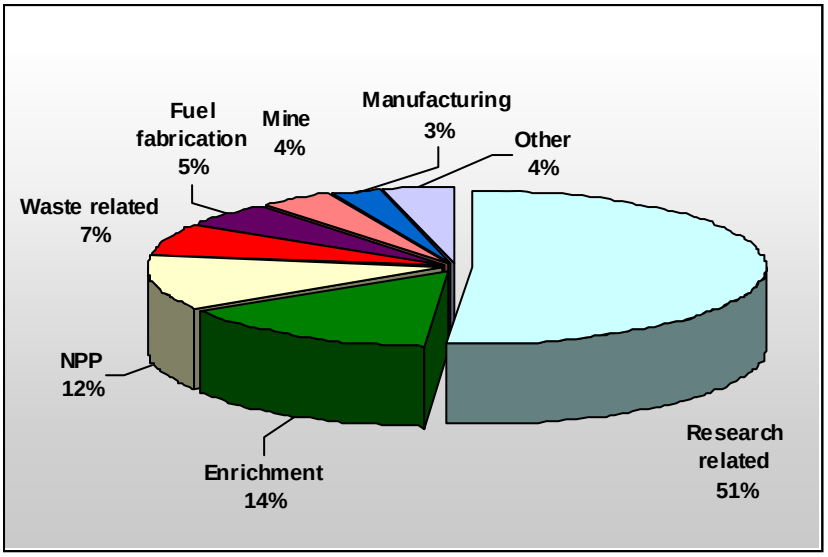


**Chart 2:** Complementary Access by type of installation.

### 3.3.2. Non-Nuclear Locations

The Agency can request access pursuant to Article 5.b. where the request concerns reports made under Articles relating to: R&D without nuclear material; the production of non-nuclear materials or equipment referred to in Annex I; the imports of materials and equipment referred to in Annex II; or locations that could be functionally related to sites, and where the activity is taking place outside a site. The minimum notice period is 24 hours, but a valid advance notice can only be given following the procedure referred to in Article 4.d. Hence before requesting access the Agency must ask the State for clarification and given an opportunity to respond. Moreover, and different from the requirements concerning nuclear locations, it is accepted that the State might not be able to provide this access. In

such case the State *shall make every reasonable effort to satisfy Agency requirements, without delay, through other means*.

There are a few locations which could be accessed based on Article 5.b., but so far no request for Complementary Access has been received.

### 3.4.3. Other Locations

Article 5.c. provides a legal basis for access to *any location specified by the Agency, other than locations referred to in paragraphs a. and b. above, to carry out location-specific environmental sampling.* The purpose of such access is *to resolve a question relating to the correctness and completeness of the information provided pursuant to Article 2, or to resolve an inconsistency relating to that information*.

Under these provisions the Agency can request access – with a 24-hour notice – to any undeclared place that raises questions about the correctness and completeness of the declarations submitted to the Agency. Before making a request for access, the Agency has to give the State (and the Community) the opportunity to *clarify and facilitate the resolution* of the question, as foreseen in Article 4.d. If we accept that Article 4 relates to procedural matters, the question could be raised under Article 2.c. as a request for amplification or clarification.

These provisions have not been used in the European Union of INFCIRC/193. This is somewhat surprising, as Article 5.c. would seem to offer a great potential for the Agency to satisfy its verification needs in a number of situations where access has been sought but referring to other legal bases. These include buildings located outside site boundaries, other locations where the Agency has raised questions about activities that might need to be declared. Article 5.c. could perhaps also be used to clarify the status of old installations that were never declared to the Agency due to their decommissioning before the entry into force of INFCIRC/193.

A reason for the reluctance of the Agency to refer to Article 5.c. might be due to the rather curious text setting out the permitted activities when Complementary Access is made under this Article. Indeed, according to Article 6.d. the only activity would be the *collection of environmental samples and,* only if the results from the environmental samples *do not resolve the question or inconsistency* would the Agency have the right to carry out visual observation and radiation detection and measurement.

It is unknown to us how these provisions ended up in the Additional Protocol. Whatever may have happened, we submit that in order for them not to be void of all practical sense, they must be read in a pragmatic way. It seems obvious that visual observation cannot be excluded if environmental samples are to be taken. Nor should it be forbidden for an inspector accessing a location which – according to his information – may be in the possession of nuclear material to make use of radiation detection and measurement devices. Perhaps the scope of activities should be limited to the purpose of access as stated in the question or request for clarification.

### 3.4.4. Complementary Access on Invitation

Article 8 provides an additional basis for Complementary Access by stating that *Nothing in this Protocol shall preclude a State from offering the Agency access to locations in addition to those referred to in Articles 5 and 9 or from requesting the Agency to conduct verification activities at a particular location.*

The Agency has expressed some interest in using this basis for access in cases where it is unclear for the State or the Agency whether a place would be covered by the reporting requirements. Article 8 would seem to offer a good basis for a regular access without the lengthy procedures related to access pursuant to Article 5.b. or 5.c. and 4.a.(ii), after a request under Article 2.c. or 2.b.(ii) has been made. Referring to Article 8 would place the 'visit' by the IAEA into the legal framework of the Additional Protocol, including the reporting requirements foreseen in Article 10. Moreover, it would be possible to lay down the purpose of access and the activities to be performed based on the necessities of the case.

In a few cases Article 8 has been under consideration for 'visits' that would have the characteristics mentioned above. In one the objective would have been to visit selected very small holders regrouped into the European catch-all material balance area, in view of clarifying their activities with respect to the reporting requirements. In another case the Agency expressed its interest in clarifying the activities of some industrial companies with respect to today's reporting requirements but also to their involvement in past research activities. So far, however, Article 8 has not been used.

Other locations where Article 8 could be helpful are some of the former installations decommissioned before the entry into force of the Safeguards Agreement. As these had never been under Agency safeguards, it is questionable whether access can be made under the Safeguards Agreement. Article 5.c. or Article 8 are potential tools, in case Agency considers that status verification is needed but not possible under Article 5.a.(iii).

Finally, the Agency has quite often requested access to locations not falling into any of the categories set out in Article 5. Requests have typically concerned places outside a declared site and the request for access has been made during a Complementary Access to the (nearby) site. In most cases access to the buildings outside the site was possible and it was provided. However, the basis of such access remains ambiguous. Was it a voluntary offer under Article 8 but without any written trace?

As a matter of principle, there can be no objection for the Agency to ask for access to any place on voluntary basis. However, there are two observations to be made with regard to such situations. The first observation concerns the manner of asking for access. For the correct implementation of the legal provisions contained in the Additional Protocol, it is important that each inspector asking for a voluntary offer clearly indicates that what he is asking for is a voluntary action. Secondly, when making this type of request, the Agency needs to be prepared to accept that no access may be granted.

We submit that the Agency should use clear legal bases in all requests for Complementary Access. Voluntary offers are perfectly in line with the letter and the spirit of the Additional Protocol, but requests indicating an incorrect basis are not. In this context it is worth pointing out that closed-down locations outside facilities without former activities related to conversion, enrichment, fuel fabrication or reprocessing and not in possession of hot cells do not need to be reported as sites. For this they do not need to be confirmed, or even declared, as decommissioned. If the Agency has queries about such places, be it because the operator possesses laser equipment, be it for other reasons, it should refrain from pretending that these places constitute 'sites'.

## 3.4. Activities

### 3.4.1. Activities Foreseen in the Additional Protocol

Article 6 provides for the activities that the Agency may conduct during a Complementary Access.

Where Complementary Access is made to nuclear sites and decommissioned installations, the Agency can use:

- Visual observation
- Environmental sampling
- Seals & other identifying and tamper indicating devices
- Radiation detection and measurement

During Complementary Access to nuclear locations such as mines, concentration plants, waste storages subject to reporting, locations with exempted material subject to reporting and locations with source material subject to reporting, the Agency can, in addition to the activities under the previous point, undertake:

- Item counting of nuclear material
- ND measurements & sampling
- Examine records on quantities: origin and disposition of material

In case Complementary Access is made to non-nuclear locations on the basis of Article 5.b. the Agency can, in addition to activities permitted for nuclear sites and decommissioning verifications,

- Examine production & shipping records.

There is little to say about the use of visual observation and radiation detection and measurement. These are the most common activities and there are no remarks to make about their implementation. Seals have been used for sealing containers with replicates of environmental samples or copies of photographs that were left on site. Samples of nuclear material have been taken in very few cases and examination of any kind of records has also been rare.

Environmental sampling is widely used by the IAEA also in Europe. 128 samples have been collected in the course of about 75% of all Complementary Accesses. The overall average number of swipes per CA was therefore 1.7, and the average for the 53 cases where swipes were actually taken 2.4. Typically one or two samples were collected during each of these Complementary Accesses. In 8 cases more than 3 swipes were taken. When comparing the first period from late 2004 to the end of 2005 to the second period running from the beginning of 2006 until mid-May 2007, we observe a reduction, as the overall average number of swipes taken per Complementary Access declined from 2 to 1.4.

As a follow-up to results of environmental samples taken during a Complementary Access, the Agency has in one case made a request for additional information concerning the operator's (past) activities. The analysis of swipe samples takes currently almost a year, which is considered very long by operators.

### 3.4.2. Photography

Photography is not foreseen in the Additional Protocol for the use of the Agency during Complementary Access. It is, however, widely seen as a beneficial means of documenting certain activities and accepted under some conditions by our Member States. As most Member States are reluctant to allow extensive use of photography during Complementary Access, an arrangement has been made to cater for the interests of both parties. This arrangement ensures that the operator has control over photographs that might be sensitive for security or commercial reasons. The operator decides which pictures are released and whether copies are given in an electronic format or on paper. The pictures that the operator considers too sensitive to be released are kept on site under seal.

Our experience shows that photography is widely used. Pictures have been taken during 53 Complementary Accesses either by the operator using his own camera or the IAEA inspector using an IAEA camera. Only in very few cases has the operator requested that pictures remain on site or be deleted due to their sensitive nature. It is important that the operator knows in advance about the IAEA's practice of requesting photos especially in large complex sites, in order to clarify before the beginning of the Access the rules concerning picture-taking in different areas. This does not happen automatically, as photography is not mentioned among the activities neither in the Additional Protocol nor in the Advance Notice given by the Agency. The operator therefore needs to be separately informed. At the latest during the meeting before the start of the Access, the rules and procedures for taking pictures need to be discussed and agreed on. It is important that the Agency respects the operator's rules on photography without exercising pressure. During the contacts we have with the Site Representative after receiving the advance notice, we explain the arrangement where it is not already known, allowing the operator to have a camera ready if the use of the Agency’s camera is not permitted.

The number of photographs has raised objections in some cases. The issue has been discussed with the Agency, with as a result the understanding that while the inspector would ask for pictures depending on his needs in each case, he should respect the principle that only necessary pictures would be taken. The number of pictures taken per Complementary Access was on average about 30.

## 3.5. Managed Access

Article 7 provides for the cases where some restrictions to the Agency's access rights are permissible. Access can be 'managed' *upon request by a State, in order to prevent the dissemination of proliferation sensitive information, to meet safety or physical protection requirements, or to protect proprietary or commercially sensitive information.* The special arrangements made for these cases *shall not preclude the Agency from conducting activities necessary to provide credible assurance of the absence of undeclared nuclear materials and activities.*

Where managed access is requested, a reference to it is included in the declaration in relation to the specific building or part of a building. This is valid for the cases where the requirement can be identified in advance. Managed access can be requested also *ad hoc*, at the start of the Complementary Access, in case the grounds for it could not be known beforehand.

Detailed arrangements for managed access were made for enrichment facilities. They were tested in practise before the start of the implementation, and they have been in use in the agreed form since then. In other installations the usual reasons for managed access are safety and physical protection.

No practical problems with the implementation of the managed access requirements have been observed. Some declarations may contain unnecessary indications of managed access. This could be the case e.g. for control rooms or offices with sensitive process data on computer screens. Even without a reference to managed access, the inspector should not seek to examine such data in detail, as the purpose of Complementary Access is to verify the absence of nuclear material and activities, and this purpose can be achieved without stepping into the room, by simply opening the door, looking in and perhaps measuring radiation levels.

### 3.6. Reporting by the Agency

Article 10 relates to the reports that the Agency is required to make to the States and the Community following its Additional Protocol activities. Pursuant to Article 10.a.(i). the Agency *shall inform the State concerned and, as appropriate, the Community of the activities carried out under this Protocol [..] within sixty days of the activities being carried out by the Agency.*

The report is a simple statement of places (buildings) accessed and activities carried out in each. It is worth mentioning that the buildings accessed on the basis of an informal voluntary offer from the operator have been included in the statement without distinction from other buildings. With regard to the timeliness of the statements, they have been sent by the Agency on average 85 days after access. Some very long delays, however, weigh on the average.

Article 10.a.(ii) asks the Agency to report on the results of activities in respect of any questions or inconsistencies brought to the attention of the State and the Community as soon as possible but in any case within thirty days of the results being established by the Agency. No experience of such reporting exists.

Article 10.b. asks the Agency to *inform the State concerned and the Community of the conclusions it has drawn from its activities under this Protocol.* These annual statements have been provided now three times: for 2004, 2005 and 2006 for those Member States where the IAEA had made Complementary Access during the year of reference. For all Complementary Accesses where environmental samples were taken, the text contains a sentence making the statement of *no undeclared material or activities* conditional on the results of the swipe analysis. This was the case for statements concerning 2005 (received in March 2006), and the same applies to the statements concerning 2006. As the statements for 2005 have not been updated following the completion of the swipe analysis, a conclusion for most Member States concerning the Agency's activities in 2005 is still missing in May 2007.

## 4. Conclusion

The Commission has acquired a solid experience of Complementary Access over the past two and a half years. This experience covers a large number of different types of nuclear and other installations. Based on this experience and our other involvement in the implementation of the Additional Protocol, we are able to contribute to harmonised implementation of the Additional Protocol throughout the countries of the European Union. This experience will serve us and the new Member States as they are currently acceding to the common Safeguards Agreement and its Additional Protocol.

Our experience has shown that, in order to ensure smooth and transparent implementation of the Additional Protocol, a regular forum for an exchange of views and clarification of doubts in concrete cases is needed for the Commission and the Agency. The authors of this paper hope that such forum

will be among the positive outcomes expected from the on-going discussions between the two Institutions.

[1] *Additional Protocol to the Agreement between the Republic of Austria, the Kingdom of Belgium, the Kingdom of Denmark, the Republic of Finland, the Federal Republic of Germany, the Hellenic Republic, Ireland, the Italian Republic, the Grand Duchy of Luxembourg, the Kingdom of the Netherlands, the Portuguese Republic, the Kingdom of Spain, the Kingdom of Sweden, the European Atomic Energy Community and the International Atomic Energy Agency in implementation of Article III(1) and (4) of the Treaty on the Non-proliferation of Nuclear weapons (1999/188/Euratom);* JO L 67, 13.3.1999, p. 1.

[2] *Agreement between the Kingdom of Belgium, the Kingdom of Denmark, the Federal Republic of Germany, Ireland, the Italian Republic, the Grand Duchy of Luxembourg, the Kingdom of the Netherlands, the European Atomic Energy Community and the International Atomic Energy Agency in implementation of Article III (1) and (4) of the Treaty on the non-proliferation of nuclear weapons (78/164/Euratom);* OJ L 51, 22.2.1978, p. 1.

[3] European Commission; *Commission Regulation (Euratom) No 302/2005 of 8 February 2005 on the application of Euratom safeguards;* OJ L 54, 28.2.2005, p. 1.

# Implementation of the IAEA's Model Safeguards Approach for Gas Centrifuge Enrichment Plants

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***Abstract:***

*The International Atomic Energy Agency has updated the model safeguards approach for gas centrifuge enrichment plants. The revised model approach emphasizes three safeguards objectives:*

- *The timely detection of the diversion of significant quantities of natural, depleted or low enriched UF6 from the declared flow through the plant, and the deterrence of such diversion by the risk of early detection;*
- *The timely detection of the misuse of the facility in order to produce undeclared product (at the normal product enrichment levels) from undeclared feed and the deterrence of such misuse by the risk of early detection;*
- *The timely detection of the misuse of the facility to produce UF6 at enrichment higher than the declared maximum, in particular HEU, and the deterrence of such misuse by the risk of early detection.*

*The new approach provides for a range of safeguards measures to meet the objectives taking into account facility-specific conditions. Among measures that enhance the basic safeguards approach established by the Hexapartite Safeguards Project in the early 1980s are:*

- *100% verification coverage of nuclear material flow by means of randomised inspections (Unannounced or short notice random inspections);*
- *Declaration of inventory changes via a "Mailbox" system;*
- *Surveillance at feed and withdrawal stations.*

*Some of the newly proposed measures are being tested at enrichment facilities in various countries. The paper describes the new elements of the updated model safeguards approach, the rationale for their implementation, and practicalities of implementation at enrichment plants.*

**Keywords:** safeguards; enrichment; mailbox; inspections; SNRI

## 1. Introduction

Commercial gas centrifuge enrichment plants (GCEP) are safeguarded by the IAEA in a number of States. The criteria for safeguarding enrichment facilities were based on the approach established by the Hexapartite Safeguards Project (HSP) in 1983, and modified in 1996 to include environmental sampling (ES).

In 2006 an updated “Model Safeguards Approach for Gas Centrifuge Enrichment Plants” [1] was developed and approved for implementation by the Agency. The approach took into account advances in technology, both at the enrichment plants and in safeguards, and the increase in size and capacity of the facilities.

The updated approach specifies three main acquisition paths that should be considered for a State having a GCEP:

a) Diversion of declared nuclear material;

b) ‘Excess production’, i.e. the introduction of undeclared nuclear material into the plant with subsequent enrichment to a level less than or equal to the declared maximum;

c) Undeclared production of high-enriched uranium (HEU) at the declared GCEP.

To cover the abovementioned acquisition paths, the model safeguards approach presents three safeguards objectives specific to the GCEP:

> Objective 1. The timely detection of the diversion of significant quantities of natural, depleted or low enriched UF6 from the declared flow through the plant, and the deterrence of such diversion by the risk of early detection;
>
> Objective 2. The timely detection of the misuse of the facility in order to produce undeclared product (at the normal product enrichment levels) from undeclared feed and the deterrence of such misuse by the risk of early detection;
>
> Objective 3. The timely detection of the misuse of the facility to produce UF6 at enrichment higher than the declared maximum, in particular HEU, and the deterrence of such misuse by the risk of early detection.

The new approach provides for a range of safeguards measures to meet the above objectives taking into account facility-specific conditions.

Among the proposed measures that enhance the basic safeguards approach established by the HSP are conducting randomised inspections, verifying nuclear material flow by flow-and-enrichment monitors, installing surveillance at feed and withdrawal stations, etc.

This paper deals with practicalities of implementation of the new elements of the model safeguards approach at GCEP, with an emphasis on application of randomised inspections.

## 2. Random interim inspections at GCEP.

Randomized inspections have been introduced for safeguarding GCEP by the HSP in the form of limited frequency unannounced access (LFUA) visits to cascade halls. Though limited in scope, the LFUAs provide an adequate tool for detection of undeclared HEU production at enrichment plants that cannot produce HEU without physical alteration of the cascade pipe configurations. However, advances in technology and substantial increase in size of the typical commercial GCEP make it a necessity to expand LFUAs by complementing them with additional measures.

The updated model safeguards approach introduces randomised inspections for verification of nuclear material flows into, out of, and within the GCEP (verification of receipts and shipments, verification of feed, product and tails) with the purpose of achieving Objective 1 and Objective 2, and contributing to achievement of Objective 3. Ideally, all information related to UF6 cylinders at the facility (date of receipt or planned shipment, related MBA name, availability for verification, cylinder ID, batch ID, gross and tare weight of the cylinder, weight of uranium total and the fissile component, enrichment, location inside the facility, including blending, feed, tail and product station ID) would be provided by the GCEP operator to the IAEA in the near-real time (daily) via remote connection. Based on the information provided, the IAEA inspectors would plan an inspection date, estimate the scope of verification activities, and conduct the inspection in unannounced mode.

Despite all the advantages of this ideal approach, its practical implementation requires addressing a set of GCEP operator concerns, related to protection of commercially sensitive information, and accommodation of nuclear material verification activities on a very short notice. In addition, a proper mechanism of notification and communication between the IAEA and the local State Authorities should be established, so that the local legal requirements would be fulfilled without hindering the IAEA ability to draw independent and accurate conclusion.

To accommodate various concerns of the GCEP operators and State Authorities, the IAEA developed and is testing a group of randomised inspections, which consists of the following:

a) Short Notice Random Inspection (SNRI);

b) Unannounced Inspection (UI);

c) Enhanced LFUA (ELFUA).

Every of the above types of inspections require submission of advanced information in the form of declarations. The most preferable way of submission is via a “Mailbox” system.

## 2.1. Mailbox system

Safeguards Mailbox is a system that is capable to automatically accept and log declarations from a facility operator without the need for having an inspector present, and store them for further processing. It consists of hardware[1], software, and an agreed framework related to hardware locations, means of communication, and declaration media. The mailbox must satisfy the following Agency requirements:

- Non-repudiation – The operator must not be able to falsely deny that the declaration was made;
- Trusted time stamp – The Agency must know when the declaration was made. The sensitivity to the accuracy of the time stamp for various facility types and declaration types depends on the agreed mailbox arrangements;
- Uniqueness – For one event, only one declaration is allowed;
- Unalterable – While procedures must be negotiated to correct errors in declarations, the operator must not be able to alter the original declaration after it is made without this alteration being obvious to the Agency;
- Secure – The infrastructure required to make the declaration must not compromise the security of the Agency’s computer or computer network.

In addition, the operator or the State might want the mailbox system to comply with some other requirements, among them:

- Noncounterfeitable – No unauthorized party can make declarations in the name of the operator;
- Reviewable – The operator must be able to perform his administrative, technical, and security reviews of the declaration before it is transmitted to the Agency;
- Secure – The infrastructure required to make the declaration must not compromise the security of the operator’s data, computer or computer network;

[1] Computers, fax-machines, telephones, etc.

- Acknowledgement – The operator should receive a time-stamped, signed receipt showing that the Agency received the declaration and accepted it. This is essential for resolving any disputes over the validity or content of the declaration.

## 2.2. Short notice random inspection (SNRI)

SNRIs at GCEP are routine inspections performed at random intervals during the Material Balance Period (MBP) with the main objective to allow complete verification coverage of the nuclear material flow into and out of the facility.

### 2.2.1. Notification

Similar to SNRIs at DNLEU[2] fuel fabrication plants [2], implementation of SNRIs at enrichment plants requires agreement with the State Authorities and the plant operator on the inspection notification procedure, including notification time, means of notification, and resolution of “force majore” situations. Currently IAEA considers the notification time for SNRI at GCEP to be 24 hours or less.

### 2.2.2. Retention

For a random inspection to have a non-zero detection probability for verification of nuclear material flow, it is required that all items declared by the operator as being available for verification should retain this condition for an agreed period of time after they are declared. This agreed period of time, often called ‘retention time’ or ‘residence time’, is facility and stratum specific, but in any case should be longer than the notification time.

### 2.2.3. Frequency of SNRI

The model safeguards approach for GCEP specifies the lowest frequency of SNRIs as a number of inspections per MBP that is sufficient to observe 20% of the items involved in the nuclear material flow, and verify them on a random sampling basis. The SNRI frequency depends strongly on the retention period. The longer the retention period, the smaller number of SNRIs per MBP are required. The retention period should also assure that each receipt or shipment of feed, product and tail cylinder has a positive probability for verification before being fed to the process or shipped from the facility.

### 2.2.4. Verification activities

During SNRI all UF6 cylinders declared by the operator as being available for verification are identified, item counted, and verified for gross, partial, and bias defect with medium detection probability, or using the optimized sampling plan. IAEA inspectors also examine facility records and reports, and compare the mailbox declarations with operational records.

In addition, the model safeguards approach provides for a set of activities during interim inspections that may, or may not, be performed during the SNRI. They include:

- Verification of cylinders attached to the process or in blending stations by item counting and weighing by the operator’s process load cells (when those can be authenticated);
- Collection of environmental samples and, where possible, samples of in-process materials;
- Application of seals;
- Evaluation of C/S measures;
- Verification of the operator’s measurement and accounting systems.

### 2.2.5. Declarations

[2] Depleted, Natural and Low-Enriched Uranium

To fulfil objective 1, every UF6 cylinder received or produced at the enrichment plant must be declared by the facility operator to the IAEA and must have a positive probability of being selected for verification by the IAEA inspectors. Declarations received before the announcement of the SNRI will be used to select items for verification.

For every declared UF6 cylinder, the declaration should contain the cylinder number, batch id, gross and tare weight, U total and U-235 fissile weight, and the enrichment. It should also specify the cylinder location, and receipt or planned shipment information.

If agreed among the State, operator, and the IAEA, other operational information, in particular status of feed, product and tail stations, may be included in the content of declarations. The arrangement preferred by the IAEA is to receive these declarations on a daily basis in a 'mailbox'.

### 2.3. Unannounced Inspection (UI)

As shown in the model safeguards approach, the main difference between the SNRI described above and the UI is that the notification time for UI is set to zero (no advance notification to the State). Due to this advantage, UI simultaneously meets the objective 1 and objective 2 requirements. In comparison, installations of additional C/S support measures may be required to meet the objective 2 if SNRI scheme is selected for a particular GCEP.

The IAEA has a right to conduct an LFUA, as well as complementary access (CA), at the time of UI or SNRI. This creates a potential additional burden on the State Authorities and the GCEP operator. Taking into account facility specific conditions, it seems logical to reduce the total number of independent randomised inspections by combining their activities.

Despite its advantages, the practical implementation of UIs raises concerns from operators, who have to support inspection activities without notice, the State, who has to arrange for a presence of national authorities at the inspected facility or delegate its responsibilities, and the IAEA. The IAEA concerns are related to difficulty of arranging a truly unannounced, surprising arrival of inspectors to the facility gate, without prior knowledge of it by the State or the operator.

### 2.3. Enhanced LFUA (ELFUA)

The enhanced LFUA, or ELFUA, is an inspection that combines features and activities of LFUA and UI. By simultaneously performing visual observation of cascade halls, feed and withdrawal stations, performing verification of nuclear material flow at GCEP, taking environmental samples, etc., this one type of inspection would allow meeting all three main objectives.

Implementing ELFUAs at enrichment plants would typically result in a larger percentage of nuclear material flow being actually observed than required by the model safeguards approach, unless the plant has a large throughput, and nuclear material strata have short residence time. For the later plant, use of ELFUAs might even reduce the total number of inspections per MBP compared to approach utilising SNRIs or UIs.

To conduct ELFUA effectively, a team with more than one IAEA inspector would have to be accommodated at the enrichment plant on the first day of inspection.

## 3. Current tasks

The updated model safeguards approach for gas centrifuge enrichment plant introduces flexibility in the process of development of facility specific safeguards approaches. It allows the IAEA to optimise available resources, and to increase efficiency of its safeguards, by selecting most appropriate set of measures to fit a particular plant. Not all the measures, provided by the model safeguards approach, are uniformly acceptable in all the Member States. The IAEA is undertaking a test of newly introduced elements of the model safeguards approach at enrichment plants in various countries. The following elements have been identified that require testing:

a) Verification of nuclear material flow at GCEP during unannounced inspection

b) Submission of mailbox declarations;

c) Specifications of the mailbox system;

d) Notification arrangements with the State;

e) Identification of cylinders attached to the cascades;

f) Definition of the number and the type of inspections in relation to the facility throughput and the retention period;

g) Use of C/S measures to increase effectiveness of SNRI;

h) Use of enrichment and flow monitors in cascades.

Upon completion, the test results will be evaluated and an optimum combination of the available safeguards measures will be selected for a particular GCEP safeguards approach, so that the IAEA safeguards would meet all three objectives.

## 3. Conclusion

The updated model safeguards approach for GCEP introduces a set of new safeguards measures to complement existing HSP approach, increase effectiveness and efficiency of the IAEA safeguards, and meet all three main safeguards objectives at a modern commercial gas-centrifuge enrichment plant. Implementation of these measures requires careful consideration of the concerns of the plant operator and the State, cooperation among all parties, testing and selecting by the IAEA of most suitable safeguards measures at a particular plant. As a final outcome, not only will the Agency ability to detect misuse of a facility and diversion of nuclear material be increased, but also, in most of the cases, the operator, the State, and the IAEA efforts will be reduced.

## 4. References

[1] International Atomic Energy Agency; *Model Safeguards Approach for Gas Centrifuge Enrichment Plants;* SG-OP-GNRL-GD-0003; 2006-06-08.
[2] International Atomic Energy Agency; *Safeguards Approaches for Depleted, Natural and Low Enriched Uranium Fabrication Plants Employing Short Notice Random Inspections: General Principles and Requirements;* SG-OP-GNRL-GD-0006; 2006-08-16.

# Nuclear Fuel Cycle Assessment of India: A Technical Study for Nuclear Cooperation

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***ABSTRACT:***

*The recent civil nuclear cooperation proposed by the Bush Administration and Government of India has heightened the necessity of assessing India's nuclear fuel cycle including nuclear materials and facilities. This agreement proposes to change decades-old-policy which aimed at preventing the spread of nuclear weapons by denying cooperation with non-NPT signatory states. After the nuclear tests carried out by India in 1998, many were convinced that India would never formally and unilaterally cap its nuclear arsenal. This state of affairs drove the desire to approach the nuclear issue through civilian nuclear cooperation. The cornerstone of any civilian nuclear technological support necessitates the separation of military and civilian facilities. A complete nuclear fuel cycle assessment of India was performed to aid in assessing how entwined the military and civilian facilities in India are as well as to move forward with the separation plan. To estimate the existing uranium reserves in India, a complete historical assessment of ore production, conversion, and processing was performed using open source information and compared to independent reports. Nuclear energy and plutonium production (reactor- and weapons-grade) was simulated using declared capacity factors and modern simulation tools. The three stage nuclear power program entities and all the components of civilian and military significance were assembled into a flow-sheet analysis to allow for a macroscopic vision of the Indian fuel cycle. These assessments included historical analysis and future projection with various possibilities of resources used.*

**Keywords:** fuel cycle assessment; India; uranium reserves

## 1. INTRODUCTION

A great deal of speculation has occurred with regards to the Indian fuel cycle (both military and civilian) since the initial agreement for nuclear cooperation between the U.S. and India was made on July 18, 2005. Much of this is moved by a misunderstanding of the technical details of the Indian fuel cycle and Indian nuclear facilities; however, some speculation is also a product of uncertainties in the status and disposition of various Indian facilities. This work is focused on the technical assessments for the Indian fuel cycle based on open source information on the Indian nuclear facilities and the usage of those facilities. Assumptions and uncertainties included in any of the models used here are explicitly declared.

## 2. INDIAN NUCLEAR FACILITIES

India has a large suite of nuclear fuel cycle facilities and reactors. The details of these facilities are described in other sources [1]. The first nuclear power project of India started with General Electric constructing and commissioning two units of Boiling Water Reactor (BWR) power plants at Tarapur in 1969. Soon India realized the difficulty in acquiring enriched uranium for these reactor types and believed that BWR's would ensure lifetime dependence on the U.S. for fuel needs. Even before India's first power plant at Tarapur was built, Homi Bhabha and his team were suggesting a three stage program and looking into the potential of CANDU reactors [4]. With this program India could make use of indigenous natural uranium for which production technology existed. This endeavour though carried the burden of acquiring heavy water for moderation and as a coolant. It led India to installing 15 operational pressurized heavy water reactors (PHWR), 3 under construction and 4 planned CANDU

power projects as of May 2007. All this however produces less than 2.5% of the electrical consumption [6]. Until the year 2005 India did not have a nuclear power producing unit greater than 220 MWe [5].

India also constructed a number of research and production reactor systems. Table 1 lists the currently operating and decommissioned non-power reactor systems in India [1]. The nuclear fuel cycle of India also includes a small enrichment facility at Mysore with a nominal capacity of 2000 SWU. Additionally, a number of facilities researching uranium enrichment methods also exist.

A large scale CANDU fuel fabrication facility is operational at the Nuclear Fuel Complex (NFC) at Hyderabad. This unit can manufacture 19 pincell fuel bundles for 6 power plants operating at 90% capacity factor. Expansion of the plant occurred recently to meet the needs of 14 PHWR's at 90% capacity factor. If India continues at the same rate of adding two power plants a year, then soon further expansion of the facility will be needed.

| NAME | LOCATION | TYPE | START DATE | FUNCTION |
|---|---|---|---|---|
| CIRUS | Trombay | 40 $MW_{th}$ HWR | 10 July 1960 | Weapon Grade Pu |
| DHRUVA | Trombay | 100 $MW_{th}$ HWR | 10 Aug 1985 | Weapon Grade Pu |
| Apsara | Trombay | 1 $MW_{th}$ LWR | 1956 | Knowledge of Nuclear Reactors |
| PURNIMA – 1 | Trombay | Critical Assembly | 1989 | Decommissioned |
| PURNIMA – 2 | Trombay | LWR | 1984 | Decommissioned |
| PURNIMA – 3 | Trombay | LWR | 1994 | Uses $U^{233}$ |
| Zerlina | Trombay | PHWR | 1961 | Decommissioned |
| Compact High Temperature Reactor | Trombay | 0.1 $MW_{th}$ Small Reactor | 2010 | Will use U-Th and $U^{233}$ to Produce Hydrogen |
| Kamini | Kalpakkam | 30 $KW_{th}$ Test Reactor | 1996 | Uses $U^{233}$ |
| Andhra University | Visakhapatnam | 0.1 $MW_{th}$ Low Power Reactor | Unknown | Planned Research |
| FBTR | Kalpakkam | 40 $MW_{th}$ Fast Breeder Test Reactor | 1998 | Prototype Fast Breeder Research and Development |

**Table 1.** Research Reactors

## 3.0 THE THREE STAGE NUCLEAR POWER PROGRAM

The importance of nuclear energy, as a sustainable energy source was recognized at the very inception of the atomic energy program in India more than five decades ago. A three stage nuclear power program (Fig. 1) based on a closed nuclear fuel cycle, was envisioned [3]. The three stage nuclear power program envisioned by Bhabha is:

**STAGE 1**: *Establishment of natural uranium fuelled, heavy water moderated and cooled PHWR for meeting electricity needs. Spent fuel from these operational reactors is to be reprocessed to separate plutonium for use in second stage reactor systems.*

**STAGE 2**: *Fast Breeder Reactors (FBR) would utilize plutonium based fuel obtained from the first stage. These FBR's breed $^{233}U$ from thorium and convert $^{238}U$ to plutonium.*

**STAGE 3**: *Advanced nuclear power systems utilizing $^{233}U$ and Thorium as fuel to provide electricity and breed more fissile content. These reactors would not only produce fuel for themselves but also excess for weapons use.*

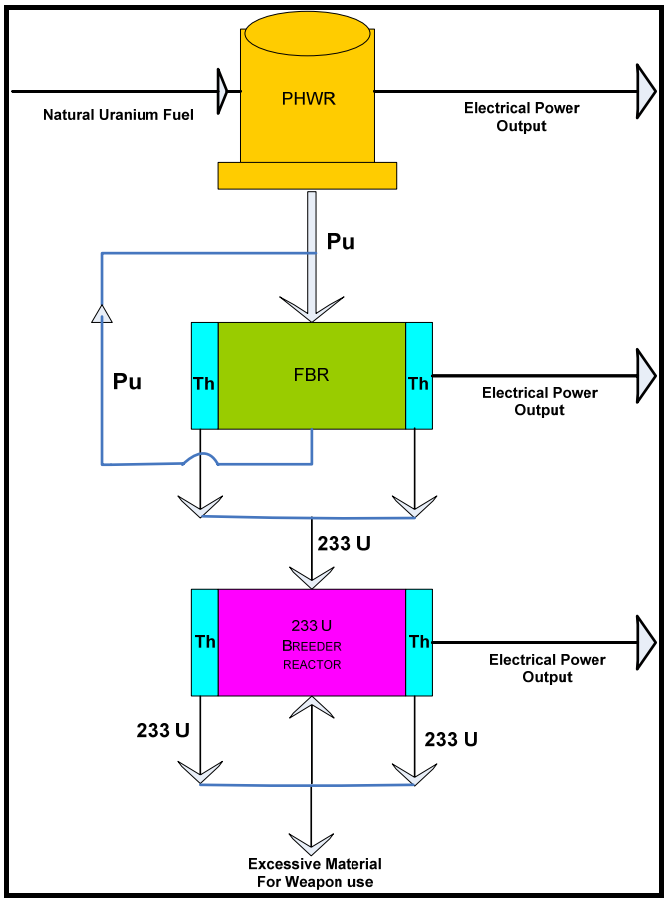


**Fig. 1.** India's three stage power production strategy

The basis of the three stage program was the indigenously available technology for production of natural uranium fuel assemblies, the vast reserves of thorium, and the mastering of heavy water production and spent fuel reprocessing technology. When this program was devised, India did not have any existing power reactors and there were no commercial fast breeder reactor systems anywhere in the world.

Over a period of time India obtained self sufficiency in PHWR technology, but until recently all of the power plants were rated at 220 MWe and ran at a low capacity factor. To advance research on development of the second stage power reactor systems, a Fast Breeder Test Reactor (FBTR) was built, based on the French Rapsodie design. It was 40 $MW_{th}$ with a mix of plutonium and uranium carbide as fuel. The design and operating experience obtained from it enabled the commencement of construction of a 500 MWe Prototype Fast Breeder Reactor (PFBR). Technological demonstration of a $^{233}U$ based reactor was done with the commissioning and operation of the 30 kW KAMINI reactor. However commercial scale systems have not yet been demonstrated.

## 4.0 FUEL CYCLE ANALYSIS TO PRESENT DAY

The fuel cycle assessment performed, accounts for the significant milestones in the Indian timeline of 1974 (first nuclear explosion), 1998 (Pokhran-II tests) and 2006 (the US-India Nuclear Cooperation Agreement). Assessment (including material production, loss and storage from all sources and facilities) was performed from inception to present day.

Figure 2 shows the nuclear fuel cycle flowsheet of India until 1974. This study concludes that by 1974 a 13.2 kg reserve of weapon-grade plutonium existed in India. Reiterating the fact that by the time India conducted the Pokhran-I test, it had the material to build only two more weapons.

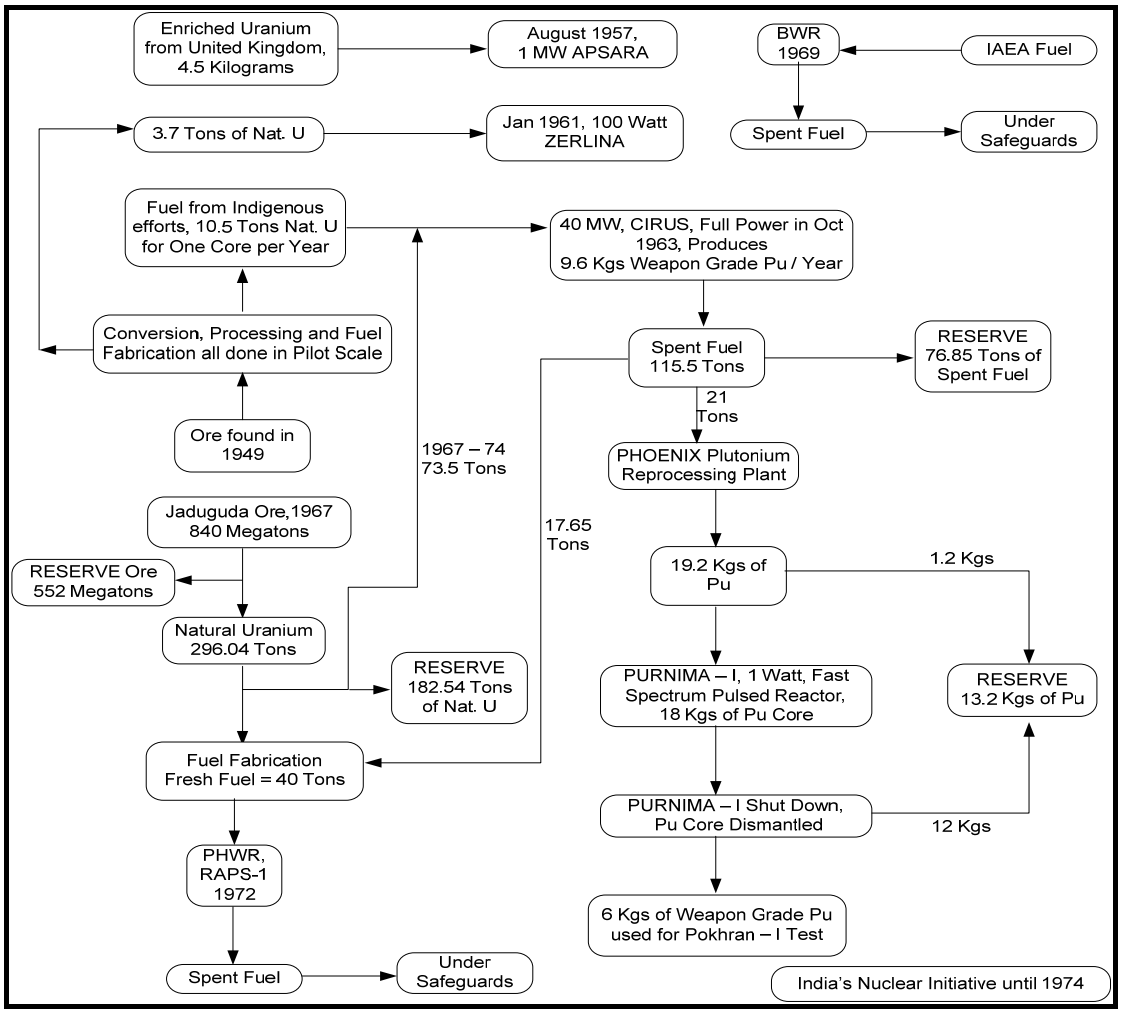


**Fig. 2.** Nuclear fuel cycle flowsheet until 1974

Figure 3 depicts the consolidated assessment of India's fuel cycle until the Pokhran-II tests in 1998. By the time of the Pokhran-II tests, India had 8 PHWR's of 220 MWe ratings and the DHRUVA reactor was producing a maximum (at 100% capacity factor) of 27.6 kgs of weapon-grade plutonium annually. An estimation of plutonium production by mid-1998 was produced from the fuel characteristics and an analysis of CIRUS and DHRUVA reactors using the ORIGEN2 and HELIOS-1.4 codes. After accounting for the weapons grade plutonium use for the Pokhran-II tests and the driver fuel for FBTR, India would have had enough plutonium for at least 44 implosion devices assuming IAEA significant quantity of 6 kgs of plutonium required for each weapon.

Immediately after the Pokhran-II tests, India was facing a dual challenge of international sanctions and diminishing uranium reserves at the flagship mine in Jaduguda. Prior to that date, the bottle neck for uranium fuel production was the milling capabilities but in the matter of a few months the focus point shifted to the uranium ore reserves. Mining activities at many other sites were attempted but

were not rigoursly pursued because of political and social reasons. In contrast, the building of nuclear power reactors increased and 6 more PHWR's were added (2 being of 540 MWe). Capacity factors of the order of 80% were achieved as of 2003 for certain power plants.

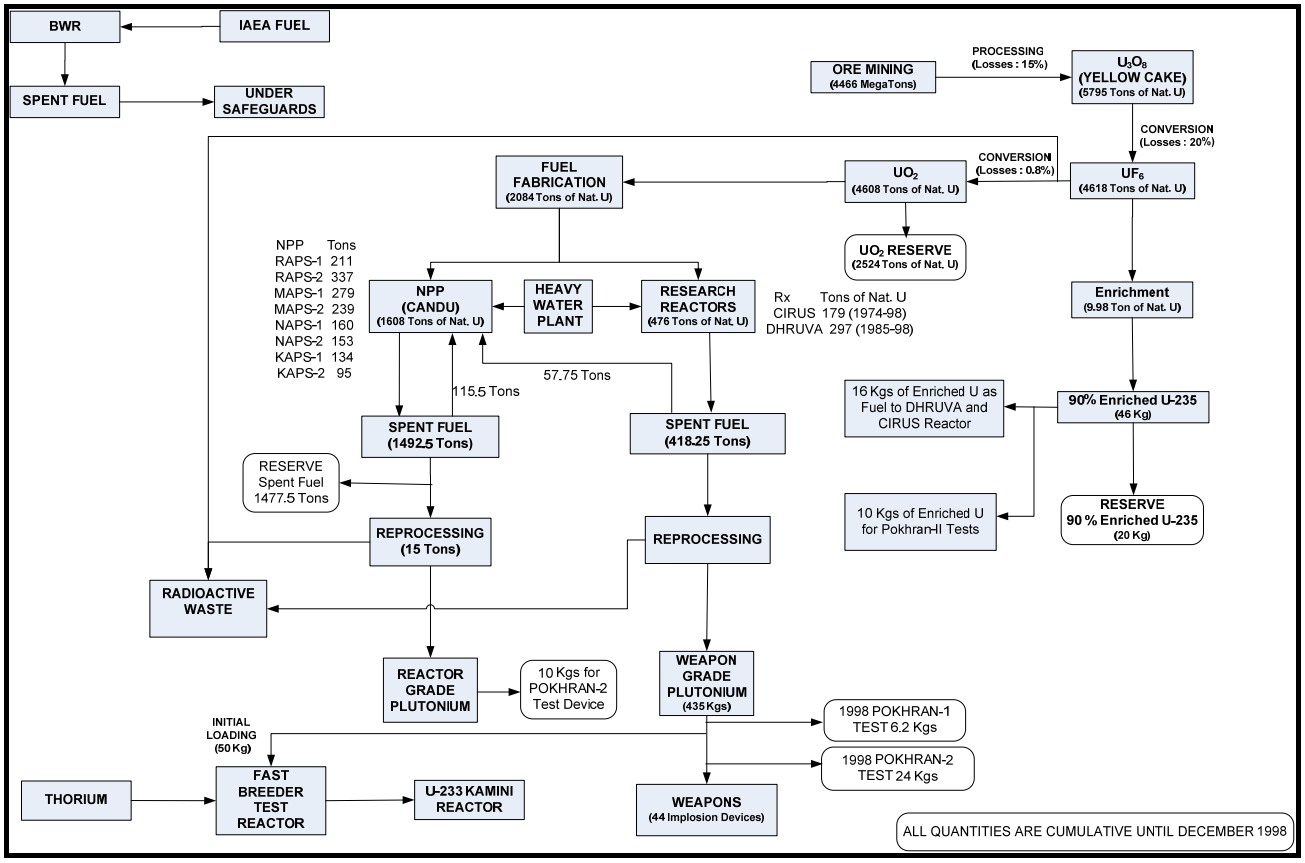


**Fig. 3.** Nuclear fuel cycle flowsheet until 1998

Analysis of uranium enrichment capabilities was performed with an assumption of P1 centrifuge machines of 3 SWU/yr capacities having a total plant load of 2000 SWU per year. India could have accumulated 94 kilograms of 90% enriched uranium by the end of 2006 after accounting for its possible use in the Pokhran-II test and as experimental fuel in DRUVA reactor. This amount of enriched uranium could fuel a nuclear submarine core if India continues in that program.

In 2006, the NFC had more than doubled its capacity. Furthermore, in 1992, two 100 tHM/yr reprocessing facilities were added. This infrastructural development shrinks the gap between the first and second stages by meeting the fuel needs of the PFBR.

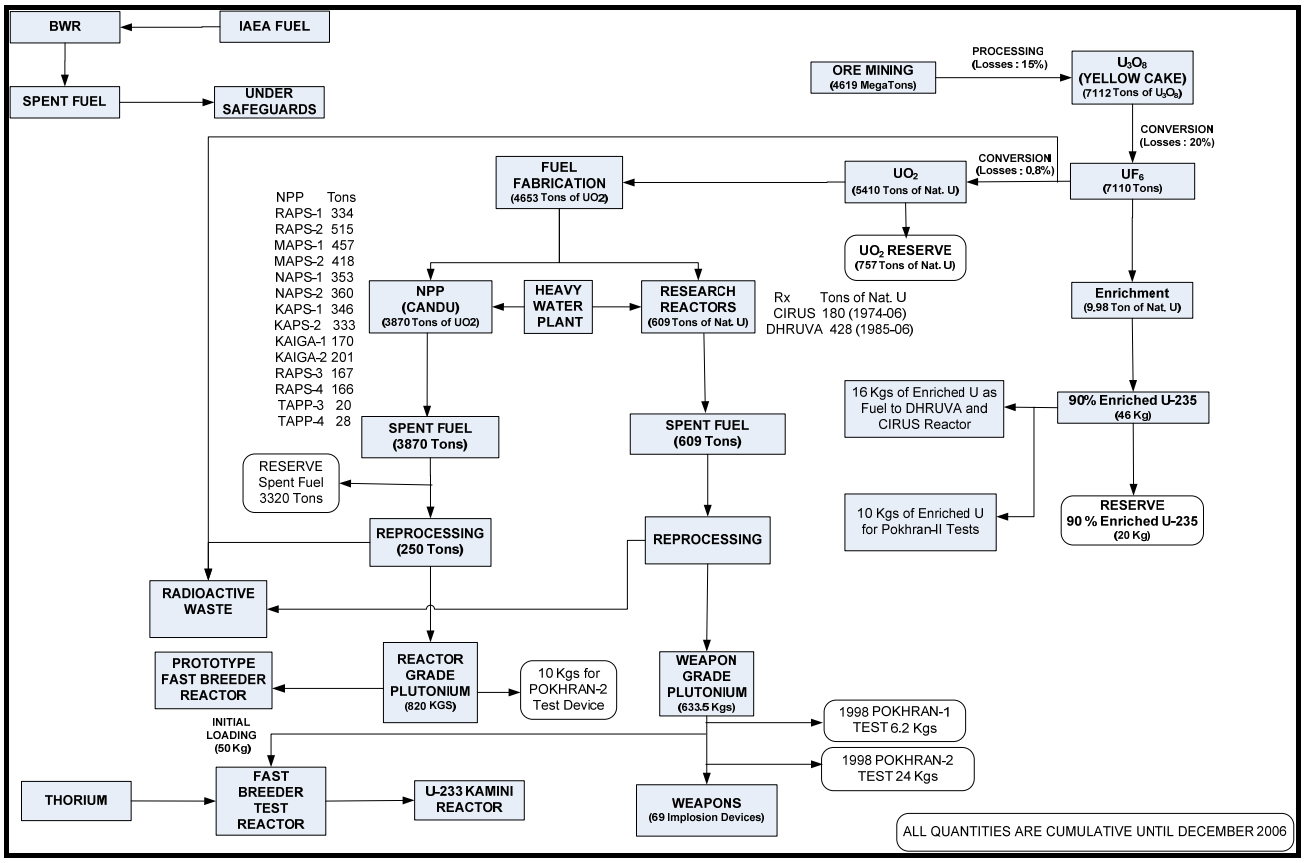


**Fig. 4.** Nuclear fuel cycle flowsheet until 2006

## 5.0 MATERIAL PRODUCTION

Exploration of uranium ores in India started as early as 1967. Beginning with Jaduguda (located in the eastern part of India), six to seven different locations were discovered over a period of time. The Jaduguda Mine had the capacity to produce up to 200 Megatons of yellowcake annually. Its actual production had been 115 Megatons per year averaged over a period of 40 years (1967-2006). For 32 years (1967-1998), the ore excavation was at the rate of 141 Megatons of yellowcake per year which dropped to an average of 10 Megatons of yellowcake per year for 1999-2006.

Mines at different locations are receiving increased attention after the exhausting of Jaduguda mines in 1999. The Narwapahar Uranium Mine became fully operational in 1999 at a cost of approximately $48.2 million. Considered one of the most modern mines in India, it has the capacity to process 7.3 Megatons/Year. The Bhatin Mine currently produces approximately 5.5 Megatons/Year.

The Mysore enrichment plant needs a feed of 2.15 tons of $UF_6$ per year to produce 10 kilograms of 90% $^{235}U$ and the Trombay plant consumes 0.43 tons of $UF_6$ every year to produce 2 kilograms of 90% $^{235}U$. Table 2 describes the quantity of materials produced per year over the stated time periods and ending with 2006. The process losses and conversions were appropriately computed for calculating the quantities of $U_3O_8$, $UF_6$ and $UO_2$.

| Time Line | Total Ore | $U_3O_8$ | $U_3O_8$ | $UF_6$ | $UO_2$ |
|---|---|---|---|---|---|
| | MEGATONS/ YEAR | TONS / YEAR | | | |
| | | | 15% LOSSES | 20% LOSSES | 0.8% LOSSES |
| 1967-86 | 141 | 254 | 216 | 217 | 165 |
| 1986-95 | 147 | 264 | 224 | 225 | 171 |
| 1995-98 | 154 | 277 | 236 | 236 | 180 |
| 1998-06 | 23 | 41 | 35 | 35 | 27 |

**Table 2.** Estimated values of $U_3O_8$, $UF_6$ and $UO_2$ annual production

The NFC has an annual handling capacity of 250 tons of yellow cake or 216 tons of $UF_6$ after losses (see Table 2). The calculated quantity of yellow cake ($U_3O_8$) was 6834 tons by 1998. Table 3 also illustrates the quantities of all the three compounds of uranium until 1998 and 2006. The scarcity of natural uranium reserves is stated as the catalyst for the India–US nuclear cooperation. Given the numerous prospective mining projects ongoing the uranium production scenario might be completely different in the future. India's ability to sustain nuclear power projects using domestic reserves may need to be reconsidered if the ore prospects are not realized.

| | $U_3O_8$ | $UF_6$ | $UO_2$ |
|---|---|---|---|
| ENDING PERIOD | 15% LOSSES | 20% LOSSES | 0.8% LOSSES |
| Until 1998 | 6834 tons | 6830 tons | 5197 tons |
| Until 2006 | 7112 tons | 7110 tons | 5410 tons |

**Table 3.** Cumulative $U_3O_8$, $UF_6$ and $UO_2$ production until 1998 and 2006

## 6.0 WEAPON GRADE PLUTONIUM PRODUCTION ASSESSMENT

The primary source of weapon grade plutonium production is from two reactors: CIRUS and DHRUVA. The thermal power rating for CIRUS and DHRUVA is 40 and 100 $MW_{th}$ respectively. Since these two do not have a declared operational history, a capacity factor of 50% and 80% is assumed respectively to compute plutonium estimates. This predicts that CIRUS reactor produces 9.6 kgs of weapon-grade plutonium per year with 10.5 tons of natural uranium fuel. DHRUVA has much shorter cycle of 67 days with 6.35 tons of natural uranium as fuel for producing 5.53 kilograms of weapon-grade plutonium per cycle. Considering a pragmatic situation of five core changes per year, DHRUVA can produce 28 kilograms of plutonium per year. Calculations of these core fuels show that total plutonium production of India by 1997 was 393 kilograms after accounting for losses in reprocessing. Extrapolating the computations with similar assumptions and inputs, the plutonium reserves would have been 633.5 kilograms by the year 2006. Table 4 shows a summary of the historical plutonium production by India.

| TIME PERIOD | WG Pu PRODUCED (KG) | NAT. U IRRADIATED (TONS) | |
|---|---|---|---|
| 1964 – 1974 | 48 | 53 | |
| 1975 – 1997 | 345 | 121 | |
| 1964 – 1997 | 393 | CIRUS / DHRUVA | 173 / 270 |
| **1964 – 2006** | **633.5** | **CIRUS / DHRUVA** | **205 / 486** |
| 2006 – 2011 | 141 | DHRUVA | 108 |

**Table 4.** Plutonium production and natural uranium use in two research reactors

## 7.0 ASSESSMENT OF POWER PRODUCTION AND URANIUM CONSUMPTION

India's nuclear power plant analysis involves assessment of fuel consumed along with spent fuel characterization for plutonium and other minor actinides recovery by reprocessing.

| POWER PLANT | % of C.F. | CRITICALITY YEAR | TONS OF $UO_2$ USED |
|---|---|---|---|
| RAPS-1 / RAPS-2 | 23.31 / 52.65 | 1972 / 1980 | 255 / 436 |
| MAPS-1 / MAPS-2 | 52.82 / 52.92 | 1983 / 1985 | 378 / 339 |
| NAPS-1 / NAPS-2 | 60.62 / 67.82 | 1989 / 1991 | 274 / 281 |
| KAPS-1 / KAPS-2 | 70.91 / 84.14 | 1992 / 1995 | 267 / 254 |
| KAIGA-1 / KAIGA2 | 80.7 / 80.91 | 2000 / 1999 | 91 / 122 |
| RAPS-3 / RAPS-4 | 77.98 / 79.2 | 2000 / 2000 | 88 / 90 |

**Table 5.** Fuel consumed by PHWR's until 2003

| POWER PLANT | C. F./YEAR | YEAR OF CRITICALITY | TONS OF $UO_2$ USED |
|---|---|---|---|
| All 12 Plants | 81% / 2004 | Operating | 366 |
| TAPP-4 + 12 Plants | 76% / 2005 | TAPP-4 on 09/2005 | 352 |
| TAPP-3, 4 + 12 plants | 52.4% / 2006 | TAPP-3 on 01/2006 | 257 |

**Table 6.** Fuel consumed by PHWR's from 2004 to 2006

The total amount of $UO_2$ produced is 5410 tons (Table 3) and the amount consumed being 4330 tons (adding up the last column of Table 6, 7) by nuclear power plants and 690 tons by research reactors (CIRUS & DHRUVA). Considering the present state of ore exploration with no additional exploration activity being added, the reserves and the meagre amount of production can last for only few years. New and bigger power plants are also under construction and would add up to the demand for fuel. Table 7 depicts the demand for fuel for 2007 from present operating plants and newer additions.

| POWER PLANT | RATED POWER | CRITICALITY YEAR | TONS OF $UO_2$ |
|---|---|---|---|
| TAPP-4 | 540 | Sep-05 | 55 |
| TAPP-3 | 540 | Jan-06 | 55 |
| KAIGA-3 | 220 | Mar-07 | 17 |
| KAIGA-4 | 220 | Sep-07 | 6 |
| RAPP-5 | 220 | Aug-07 | 6 |
| RAPP-6 | 220 | Sep-08 | 0 |
| 11 Operating | | | 249 |

**Table 7.** Fuel to be consumed by PHWR's at 60% capacity factor for the Year 2007

All the NPP's in India are presently operating at 60% or lower capacity factor. The same is assumed for all the under construction power plants that may line up at the projected dates. Recently the NFC handling capacity was increased from 250 tons of $UF_6$ to 600 tons of $UF_6$ per year to take care of the demand for producing 450 tons of $UO_2$ annually considering 14 power plants operating at 92% capacity factor. [600 tons of $UF_6$ per year would lead to 458 tons of $UO_2$ per year]

By December of 2007 India would consume 388 tons of $UO_2$ (operating all the 16 PHWR's at 60% capacity factor). If the operating capacity factors are maintained, then with the additions of newly built CANDU power plants, 397 tons of $UO_2$ will be consumed by end of 2008. This makes the total fuel to be used in its lifetime equal to 4835 tons. Now the amount of $UO_2$ produced after subtracting the $UO_2$

consumed by plutonium production reactors is 4833 tons. This evens out the production and consumption. The fuel exchequer thus goes to zero by the end of December 2008.

## 8.0 SPENT FUEL ANALYSIS

As can be inferred from the flowsheet representation of the fuel cycle, most of the spent fuel of the CANDU reactors is available for reprocessing. For first six cores depleted bundles were loaded for flux flattening. Later thorium bundles were used for flux flattening in fresh cores. A 220 MWe power plant, operating at 100% capacity factor needs eight 19-pincell fuel bundle replacements every day.[2] After correcting for the spent fuel loading for the fresh cores, actual capacity factor of operations and expecting that none has ever been sent to reprocessing facilities, 5020 tons of depleted uranium is assumed to be in the spent fuel bays (4330 tons from NPP's and 690 tons from RR's).

Taking into account an average burnup of 6500 MWd/tU for CANDU fuel bundles, 4.1 kgs of plutonium (in the ratio of $^{240}Pu/^{239}Pu$ = 42%) can be extracted from one ton of spent fuel. Given the fully operational reprocessing capability of 50 tons of spent fuel per year, India can extract 205 kilograms of plutonium every year from the spent fuel. This is appropriately sufficient to support the second stage of nuclear energy comprising of FBR's. If the two semi-operational reprocessing plants of 100 tons each become fully operational then a 500 MWe FBR can be added every year to the power grid.

## 9.0 CONCLUSIONS

Presented was a flowsheet assessment of the Indian nuclear fuel cycle. This assessment shows that without additional uranium mines being discovered in India, domestic uranium production will not be able to support the power reactor program beyond 2008. The weapons program however does not seem to be deterred by the present status. The fuel needs of CIRUS and DHRUVA can be met with the domestic production of uranium. The plutonium production reactors can still continue to operate with the meagre annual production and the separations capacity is adequate to continue weapons production at the current rate.

International supplies of uranium would allow India to continue the civilian energy production program indefinitely. Again, however, this has little effect on the weapons program. It remains likely that the weapons program will continue at the present rate irrespective of the status of the U.S.-India Nuclear Cooperation Agreement.

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# Session 17

# Remote monitoring

# Remote Data Transmission for safeguards verification purposes

# First experiences with the Sellafield MOX plant

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***Abstract***

*The new safeguards approach of the European Commission aims to improve the efficiency and effectiveness of inspections. One of the aims is to use modern technology in order to focus safeguards activities at the sites to the tasks that require inspectors' presence. If the information required to perform safeguards verifications is remotely available, the related evaluation activities will be performed at headquarters. These activities will contribute to the overall safeguards conclusions and allow the inspectors to focus on-site activities on in-process verifications, verification of the basic technical characteristics, issue follow up and other safeguards activities requiring physical presence of inspectors.*

*Remote data transmission (RDT) is being realised from the Sellafield MOX plant (SMP) in UK. This achievement will be possible because of the close co-operation with the operator and the British government authorities.*

*At the Sellafield site a new, secured computer infrastructure has been installed to centralise data for transmission to the Commission Headquarters in Luxembourg. Of highest importance is the protection of the data involved. High level encryption devices, accredited by the British government authorities, are used to create a virtual private network between the two locations and allow the safeguards' inspectors remote access to instrument and surveillance data. In addition state of health and set up checks are possible to improve the planning of maintenance interventions.*

*This presentation will discuss some of the expected benefits of remote data transmission for highly automated plants and the effects on safeguards inspections.*

**Keywords:** nuclear safeguards; remote data transmission, data security

## 1. Introduction

Nuclear installations under EURATOM safeguards are built and operated all over Europe. Safeguards activities at the installations have to be not only effective but also efficient. Therefore, when sending inspectors to these places to perform verifications, they should be focused on activities that require human presence and can only be done at the installations. The workload overhead for travelling is substantial and manpower is better spent at headquarters working on transmitted data. In addition this concept allows inspectors to perform their evaluations without the onsite constraints like time restrictions, operator, material and equipment availability, as well as the limited technical support.

Details on nuclear material quantities, locations and movements are considered sensitive in all member States. These security concerns have to be addressed when transferring data remotely. The safe transfer of the above data is therefore of utmost importance. Existing systems make use of high level encryption devices, like the Datacryptor, which are up to defence standards. VPN solutions provide the best option.

## 2. Required data

The data required for remote data transmission will be in principle the same as available on site. For the operator there will be no apparent changes in what concerns the type of data to be provided. The inspection routines will be similar as if inspectors are on site for inspection. Post box systems might be of help to standardise data exchange. Virtual private networks might be an option to work with one set of data and to simplify the data exchange.

The operator's data will involve all what is required for complete verifications as during on-site inspections: accountancy records, operating records and supporting documentation. In most of the modern plants all these data are available in an electronic format anyway; therefore no additional preparation for transmission would be required.

To be able to check accountancy declarations against the physical reality the inspectors also need information for independent verifications through access to data provided from unattended safeguards instrumentation and surveillance systems installed. The complete set of instrument data will also be transmitted and includes raw data files, alarm logs and instrumentation set up files.

## 3. Remote monitoring test at SMP

Because of its modern layout a decision was taken between BNFL and DG TREN to choose SMP as a test bed to implement a system of remote monitoring in the safeguards field [1].

For this test the first type of information that was transmitted was 'state of health' information on safeguards instruments. These data indicate if, for example, an alarm, a warning or an error was triggered by the Data Acquisition Module of the RADAR/CRISP system [2], which is used to record and evaluate safeguards signals. These data consist of log and alarm files. The advantage of transmitting these data is that a better preparation and timely action to rectify faults is possible, thus avoiding or minimising the need for re-verifications.

The information transmitted is in the form of simple text messages, allowing DG TREN technicians and inspectors to check whether the instrumentation is running or to establish where possible error sources or technical problems exist. These messages have proven to be extremely useful in monitoring the performance of the equipment in the field. Based upon the information received DG TREN has been able to plan maintenance visits on short notice. This avoided undue lengthy periods between the identification of the fault and the subsequent action to rectify the problem, which might have had expensive consequences to re-establish safeguards continuity of knowledge.

The information that is being transmitted is not classified and the DG TREN network is 'air gapped' at both ends from the BNFL and the DG TREN internal networks resulting in a minimal security risk for both the operator and the inspectorate.

These tests were very successful and it was decided to extend the data transmission to cover measurement and signal data which are more sensitive information. This has greater benefits in terms of increasing the efficiency and/or reducing the costs and intrusiveness of inspections whilst still providing the required levels of assurance to the safeguards authorities.

## 4. Further network implementations for RDT

During 2006 the DG TREN network based in Sellafield was set up for complete remote data transmission (see figure 1).

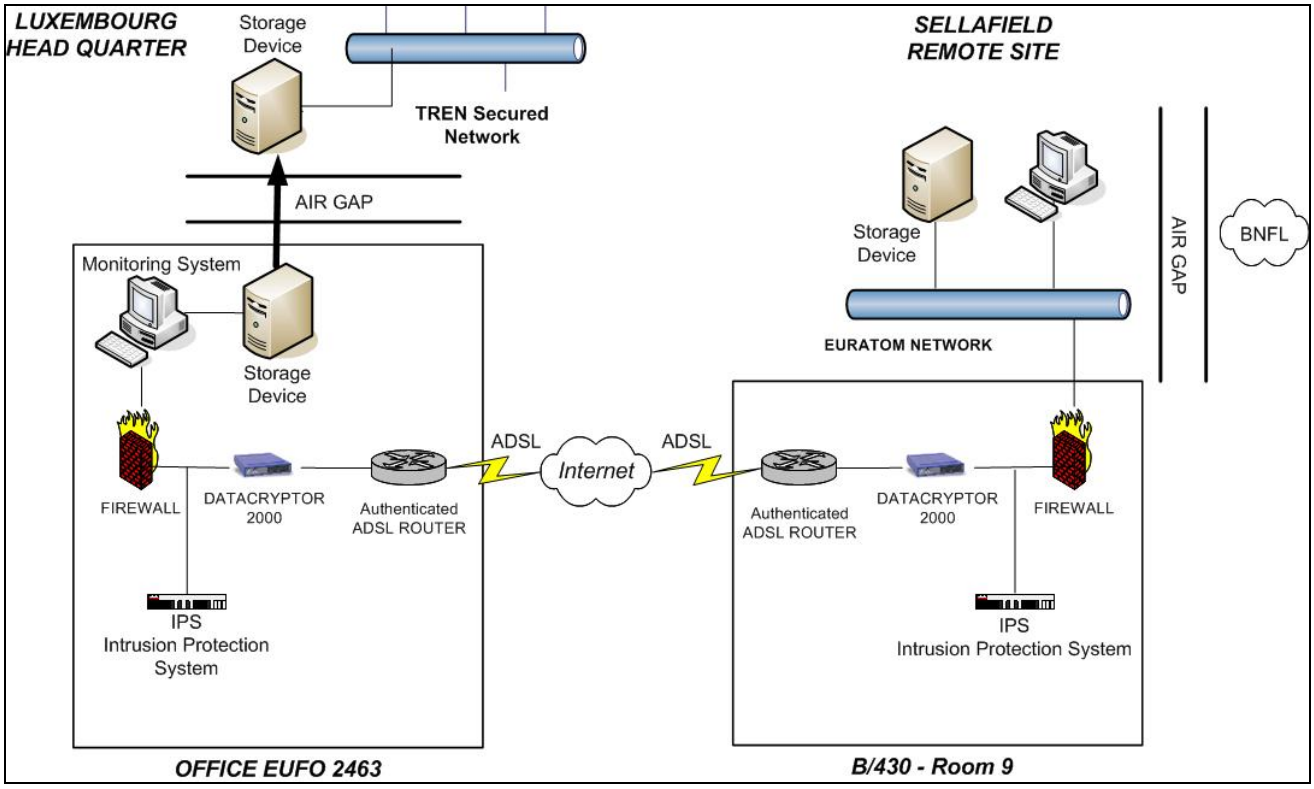


Fig. 1 Networks layout

The electronic data collected at Sellafield are held on a local area network isolated from the operating system of the main plant. All instrument data required for evaluation and assessment purposes will be passed through the Internet to an office at the DG TREN headquarters in Luxembourg using a VPN. The secured tunnel created with Datacryptor™ 2000 units will only allow communications between DG TREN headquarters in Luxembourg and the network in the on-site inspectors' office. The integrity of data will be ensured by the different components of the system which will prevent all external interactions. For the accreditation process in was decided to use the same process of UK security accreditation for the DG TREN transmission system both in the UK and Luxembourg as is applied for the BNFL network. The interim approval to operate the system was given in September 2004. Full accreditation is still to be granted.

Due to this long accreditation process of both networks, so far only tests have been carried out with the systems installed but the data transmission is at this stage not fully operational.

## 5. Relevance of RDT for the inspection approach at the Sellafield MOX Plant (SMP)

The inspection activities performed during routine inspections at SMP include amongst others accountancy verifications and physical verifications. Physical verifications are a combination of unattended measurements of items with the RADAR/CRISP system at strategic flow locations in the plant and additional random checks of items in process areas. For continuity of knowledge purposes, surveillance review with means of a FAST system is also carried out.

For the unattended measurements two basic sets of information are needed, the declaration on the movements of material through the process from the operator and data from the in-plant safeguards instrumentation. Both sets of information are available in electronic format for automatic verifications.

The operator's declarations on material moves in the process are handled by software applications which check them for internal consistency and allow querying, structure and interfacing the data provided.

Dedicated safeguards instruments running in unattended mode and signals branched from the operator's systems provide real-time information to the inspectors' site offices. All signals and raw data are treated by the RADAR/CRISP application. The combination of signals at strategic points allows for the detection of events. Events are a combination of signals that are created when items pass through process locations with safeguards

instrumentation installed. An event is for example a combination of time stamped signals from a neutron detector with a gamma system and a related bar code reader. The combination of these signals allows calculating the flow of material quantities that need to tie in with the operator's declaration.

The flow verifications consist of the subsequent comparison of the consistency checked operator data with the events detected by RADAR/CRISP. At present DG TREN is working on a standardised inspection software application called VARO. The aim is to check and validate operators' data with a standardised application and to have an automatic information exchange with the measurement applications, mostly RADAR/CRISP.

Figure 2 shows what will be in future the final implementation of the software and hardware tools in SMP integrating both, the use of automatic verification tools and remote data transmission.

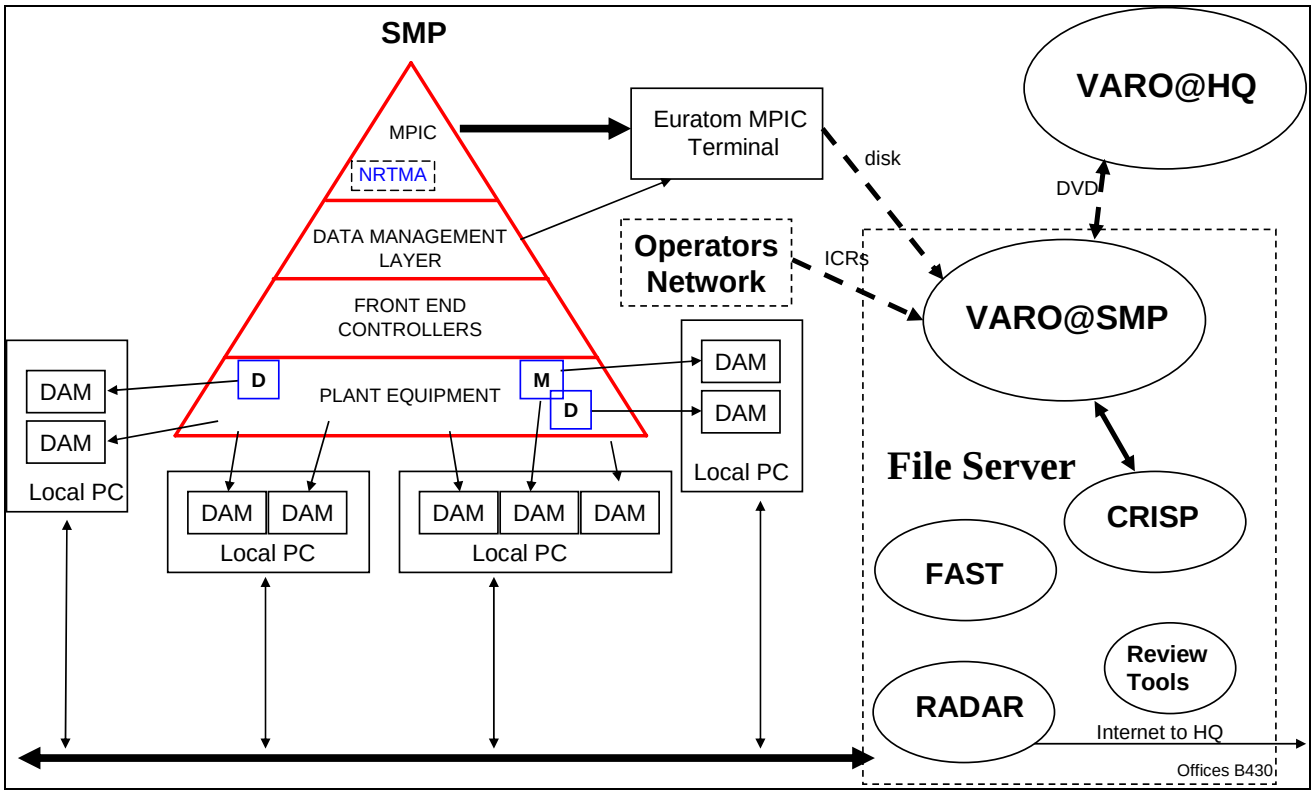


Fig. 2 Implementation at Sellafield DG TREN offices

If all information described above is available at headquarters, the inspection activities could be performed independently on-site and in headquarters leaving the decision to DG TREN on how this balance is to be implemented.

## 6. Advantages of RDT

The availability of remotely transmitted safeguards data will allow the inspectors to better prepare for inspections. They will be able to carry out the data evaluation at headquarters in preparation for planned inspections and arrive at the installations with a list of issues to be clarified. This will also allow being more specific in discussions with the operator.

The availability of instrument data at headquarters will allow for a better preparation of planned interventions or breakdown maintenance. The inspectors and/or technicians will better be able to prepare on site activities by checking the instrument data and state-of-health messages available. A further development could be the remote control of instrument parameters, which is not foreseen so far, but could be a useful tool to further minimise on site interventions.

The availability of data from the site at headquarters allows the use of common review tools. Applications like RADAR/CRISP [2] already have the functionalities to deal with data from different installations and other tools like VARO are in development.

Centralising evaluation routines will make it easier to use and maintain software applications without having to send technicians for maintenance or to use inspection resources for these interventions. All software upgrades can be first tested at headquarters and then implemented on site.

One of the major challenges in safeguards is the training of inspectors on the variety of tools used. The availability of site data at headquarters allows for the training of inspectors with real data instead of faked set ups that often only give a blurred picture of the situation at the installations. Moreover, with the availability of inspection software tools at headquarters, the use of standardised evaluation routines and criteria will be more easily implemented and generic software applications developed consequently. This harmonised approach will allow for a more efficient inspection activities and training of inspectors.

Finally, one of the common goals of any inspection based organisation is the optimisation of the use of manpower. Sending an inspector to an installation is causing a considerable overhead both in terms of travel costs and time. All evaluations that can be done at headquarters have in addition to the saved travel overhead the advantage that the inspector is not under time pressure to finish the verifications in the timeframe of the plant visit. This leads to more flexible arrangements and possibly also to better overall inspection results.

As a compromise for installations or countries having a problem with Remote Data Transmission of sensitive information a transfer of the relevant safeguards data on mobile media like CD, DVD etc. by the inspectors themselves could help to establish the necessary routines for a remote data transmission in the long run.

## 7. Conclusion

Successful and comprehensive safeguards have always been built on co-operation of all parties involved. Remote Data Transmission of sensitive safeguards information and signals is another example of how result oriented co-operation can help to make safeguards inspections more efficient and effective without compromising the quality of the evaluations.

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# Session 18

# NDA IV (spent fuel and calorimetry)

# Survey of Seven Measurement Techniques for Quantifying the Fissile Content of Spent Fuel

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***Abstract:***

The motivations for quantifying the fissile material in spent fuel are increasing. Given the potential safety, security, and energy benefits of recycling spent nuclear fuel, it is likely this fuel will be moved in greater volume throughout the world in order to be reprocessed or moved to long-term storage. This mass movement of spent fuel motivates improving shipper/receiver measurements. Furthermore, in the context of pyrochemical reprocessing, the absence of an accountability tank measurement increases the need for quantifying the fissile content of material entering such a facility as accurately as possible.

This paper will survey seven measurement techniques for quantifying the fissile content in spent fuel. In some cases, Monte Carlo calculations or another analysis will be performed to quantify key issues such as the intensity of active sources needed. The seven techniques to be studied are delayed-neutron counting (with various interrogating sources), differential die-away, lead slowing-down spectrometry, neutron resonance absorption, passive multiplicity counting, passive neutron albedo reactivity, and x-ray fluorescence.

**Keywords:** spent fuel; measurement; plutonium content

## 1. Introduction

The motivation for directly measuring the fissile content of spent nuclear fuel is driven by safeguards interests. The types of measurements and the analysis employed are determined by the specific interest of the regulatory body involved. In the context of this work, the interests of three regulatory bodies are considered: two domestic to the United States, the Nuclear Regulatory Commission (NRC) and the Department of Energy (DOE), and one international, the International Atomic Energy Agency (IAEA). Furthermore, since some of the motivations for measuring spent fuel involve concerns that presently do not have clear regulation, one of the goals of this work is to start down the path of indicating what is possible in order to better inform regulatory and policy-making bodies.

Five motivations for directly measuring the fissile content in spent nuclear fuel are described below:

(1) To determine the input accountability value in a pyrochemical processing facility. The design of the Advanced Fuel Cycle Facility (AFCF) includes both pyrochemical processing and aqueous reprocessing. Since an input accountability tank would be very costly to include in a pyrochemical facility, a key research-and-development need identified by the AFCF Safeguards Team is to quantify the plutonium content (and possibly americium and neptunium) in spent fuel.

(2) To independently verify the plutonium content of spent fuel, as requested by the IAEA. Spent-fuel measurement techniques would allow the IAEA to recover continuity of knowledge as well

as to possibly detect protracted diversion.

(3) To assure regulators that all the nuclear material of interest leaving one nuclear facility arrives at another nuclear facility ("shipper/receiver"). Given the large stockpile of nuclear fuel at reactor sites around the world, this fuel will need to be moved to either processing facilities or long-term storage sites. Safeguarding this transportation is of significant interest.

(4) To find a measurement solution for ABR reactor fresh fuel. The conditions for measuring the start-up *fresh fuel* presently proposed for the advanced burner reactors (ABRs) under the Global Nuclear Energy Partnership (GNEP) are close to the same measurement conditions that are expected for measuring *spent fuel* from LWRs. The present ABR fuel design includes all the transuranic actinides from the LWR spent fuel. The only major difference between fresh ABR fuel and spent LWR fuel is a higher gamma dose in the LWR; this difference will require lead shielding. Since lead shielding should not significantly alter the performance of neutron diagnostics, finding a measurement solution for LWR spent fuel will assist in measuring ABR fresh fuel.

(5) To enable the NRC or DOE regulations to measure LWR and research-reactor spent fuel that is not considered "self-protecting." Because of the consideration presently being given to increasing the dose level at which fuel is considered to be self-protecting and because the stocks of spent fuel are continually aging, it is likely that the quantity of spent fuel that does not qualify as self-protecting will grow. New measurement techniques can enable safeguarding of this material.

## 2. Measurement Ramifications of Different Regulatory Bodies

### 2.1 What is to be measured?

The measurement requirements for each of the spent-fuel motivations listed above are not necessarily the same. Quantifying plutonium content is the most desirable goal, but it may be sufficient in some cases (such as with shipper/receiver activities) to determine a unique nuclear signature. Some examples of possible nuclear signatures could include fissile content, total neutrons, or the neutron multiplicity distribution. Because the IAEA presently relies primarily on an indirect signature (Cerenkov radiation) for safeguarding spent fuel, some of the nuclear signatures just listed may also assist the IAEA. Quantifying the amount of uranium, neptunium, and americium may also be of interest to future regulatory activities.

### 2.2 Measurement ramifications of different regulatory bodies

Although proliferation by a nation (State) is of primary interest to the IAEA, whereas domestic regulators are primarily concerned with insider proliferation, the types of measurements and analysis acceptable to both parties are different. In the context of spent-fuel measurements, an important ramification of this difference is the acceptance of burnup-code results. Since burnup codes require input from the facility (State)—such as how long and where the fuel was in a reactor, power levels, cooling time, etc.,—the results from burnup codes do not provide verification that is independent of the facility or State. Therefore, the IAEA does not accept burnup-code results. Domestic regulators may accept such input and may use burnup codes. In the interest of being more generally applicable, this work proposes the use of measurement-based calibration techniques instead of burnup codes. The type of data envisioned with there measurement-based calibration techniques are the same as obtained with the FORK[1] or SMOPY[2] detectors.

Some other ramifications of the differing views of the facility (State) encompassed in the mandates of the IAEA and domestic regulators, in the context of spent-fuel measurements, are the following: (1) The IAEA would prefer instruments that are easily portable because their inspectors must go to many sites for a few days at a time. The facilities are required to maintain their own accountability systems, and the IAEA must check these accountability books. (2) Again, since IAEA inspectors go to many facilities and have a short time interval for making measurements, for speed and convenience reasons, they would prefer a technique that measures spent fuel in the pool rather than dry. A large nuclear facility can more readily accommodate measurements made in the air. (3) Furthermore, a large reprocessing facility may choose to measure individual spent-fuel rods or baskets of chopped rods instead of

entire assemblies, for the sake of accuracy. The IAEA would prefer a technique that is applicable at many sites.

## 3. Description of Neutron Source Term from Spent Fuel

A key quantity to determine, in the context of spent-fuel measurements, is the intensity of the radiation emanating from the spent fuel. Since most of the proposed measurement techniques are neutron based, the neutron intensity is of particular interest. Starting with LWR fuel,[3] the total neutron production from a typical $UO_2$ spent-fuel assembly that comes from a pressurized water reactor (PWR) is ~3 × $10^8$ n/s for a 5-year cooling time. If mixed oxide (MOX) fuel was used instead of $UO_2$, the intensity would be ~4 × $10^9$ n/s. In a boiling water reactor (BWR), the intensities of both of the previously quoted values would fall by approximately a factor of 3. Since the length of the assemblies for the above examples varied from 3.3 to 3.8 meters, the intensities of the LWR spent fuel per meter ranged from ~3 × $10^7$ n/s for $UO_2$ BWR fuel to ~1 × $10^9$ n/s for MOX PWR fuel. In all the above cases, allowing the fuel to cool for a total of 30 years reduces the neutron intensity by a factor of 2.5 compared to the 5-year cooling times assumed above.

Estimating the neutron source intensity for a spent ABR assembly is challenging since the design of an ABR fuel is a topic of active research. Furthermore, the proposed fuel composition for the ABR is significantly different than that from past fast reactor experiments since the primary goal of the reactor will be to burn actinides rather than to breed plutonium. The case for which the source term is estimated in this paper is the equilibrium-fuel-cycle metal fuel.[4] The term equilibrium refers to the value obtained after the actinides in the ABR fuel have been recycled back into ABR fuel and burned several times. The equilibrium value is more intense than the start-up value. The percentage of heavy metals composed of transuranic elements varied from 14% to 98% in the ABR fuel design study used for estimating the neutron intensity. Over this transuranic range, the neutron intensity varied from ~4 × $10^8$ n/s to ~3 × $10^9$ n/s for a 1-meter section. Per unit length, comparing the most intense case for LWR and ABR spent fuel, the ABR fuel is a factor of 3 more intense in neutron emission. It is worth noting that elements with atomic numbers greater than that of curium were not quantified in the publication used to estimate the neutron source strength. Even though the ABR is being designed to burn up actinides, the presence of such an elevated level of actinides in the fuel may create large quantities of higher actinides such as californium. For the present work, it is assumed that the presence of californium is negligible or that a longer cooling time can render californium negligible. A further assumption is made that the actinide content of fresh equilibrium ABR fuel is about the same as spent equilibrium ABR fuel. The justification for this assumption is that this fuel is at an equilibrium state at which the actinides burn up at the same rate that they are added in the fuel fabrication process.

## 4. Measurement Techniques

In the following seven subsections, seven measurement concepts are described that may provide information useful for determining the plutonium content in spent fuel. It is likely that several techniques, including some not mentioned in this paper, will need to be combined to achieve this goal.

### 4.1. Delayed-neutron counting

#### 4.1.1 Concept

When a nucleus fissions, approximately 99% of all the neutrons liberated from this fission are emitted at essentially the same time as the fission; the remaining approximately 1% of the neutrons are emitted by the fission fragments later in time. The fission fragments emitting the delayed neutrons are often categorized in groups by half-life; the half-lives range from a fraction of a second to just under a minute. The delayed-neutron intensity per induced fission varies by isotope. Of particular interest in the context of spent fuel, the fission fragments from the thermally induced fission of $^{235}U$ produces approximately 2.6 times more delayed neutrons than the fission fragments from the induced fission of $^{239}Pu$.

In order to perform delayed-neutron counting, a strong neutron source located close to the item of interest is turned on to induce fissions and is then turned off so that delayed neutrons can be measured. The number of delayed neutrons emitted, as the result of an induced fission in a given fissile or fertile isotope, is proportional to the mass of that isotope. In order to determine plutonium content, the

relative quantities of fissile and fertile isotopes are needed.

A positive attribute of the delayed-neutron measurement technique, relative to several of the others techniques described in this paper, is that it operates at relatively elevated neutron energies; therefore, self-shielding is low. The walls of the instrument are lined with cadmium, which limits the neutron energy returning from the walls to energies greater than approximately 1 eV. Shuffler experience has indicated that for most measurement conditions the average energy of neutrons inducing fissions are ~1 keV.[5]

### 4.1.2 Design options

For more than 30 years, the delayed-neutron-measurement concept has been used in an instrument known as the shuffler.[5] A wide range of measurement items, from solid and liquid waste to large samples of uranium to weapons pits, have been measured in the shuffler. The shuffler gets it name from the fact that a strong californium source is moved into and out of the measurement chamber to allow for irradiation and delayed-neutron counting. A clear challenge in applying this concept to LWR and ABR spent-fuel measurements is the intensity of the required interrogating neutron source, since the delayed-neutron intensity must be comparable to the neutron intensity coming from the spent fuel.

Fig. 1. The spent naval fuel shuffler is depicted above. A container is suspended above a small tube where assays are done on waste material inside the container. A larger tube, to the left, is for fuel assemblies.

Combining the results from the LWR and ABR spent fuel provides a neutron source strength ranging from ~3 × $10^7$ n/s to ~3 × $10^9$ n/s per meter. The next step is to estimate the intensity of the interrogating neutron source strength needed to create a delayed-neutron intensity similar in magnitude to that of the spent fuel. Two approaches are described in the following two paragraphs for estimating the intensity of the interrogating neutron source strength. In the first approach, the delayed-neutron count rate is estimated from expected detector parameters. In the second approach, the design properties of a shuffler used to measure highly enriched naval reactor spent fuel are scaled up to estimate the interrogating source strength needed if the spent fuel emits ~3 × $10^9$ n/s per meter.

The probability of a source neutron inducing a fission is approximately 1%. For each fission, approximately 1% of the neutrons released are delayed. If approximately 20% of the delayed neutrons produced are counted, it takes 50,000 interrogating neutrons to get one delayed-neutron count. Therefore, in order to have a delayed-neutron signal that is 20% of ~3 × $10^9$ n/s, a source with an intensity of ~3 ×

$10^{13}$ n/s is needed. This instantly rules out commercially available californium sources that are generally limited to 10 mg, which corresponds to an ~2 × $10^{10}$ n/s intensity.

The second approach for estimating interrogating source strength is the following: A shuffler was designed to measure highly enriched spent naval reactor fuel at the Fluorinel Dissolution and Fuel Storage Facility (FAST) located at what is presently known as Idaho National Laboratory.[6, 7] The requirements were for a one-sigma precision of ±2.5% over a $^{235}$U mass range of 2 to 10 kg, for a background of 5 × $10^5$ n/s from an assembly that is approximately 1.5 m long. It was shown that an ~1 × $10^9$ n/s interrogating source sufficed. Assuming that the same ratio is needed between the interrogating neutron source strength and the spent-fuel neutron intensity, then the ~3 × $10^9$ n/s neutron intensity case indicates that a ~0.6 × $10^{13}$ n/s interrogating source is needed. Note that 1 meter of ABR fuel will contain approximately 10 kg of $^{239}$Pu. From a delayed-neutron perspective, this is approximately 4 kg of $^{235}$U; therefore, ~0.6 × $10^{13}$ n/s should suffice for measuring the ABR equilibrium-fuel-cycle metal fuel because the mass range of the FAST shuffler operation was from 2 to 10 kg. Assuming that 75% of the $^{239}$Pu is burnup in the ABR, there will only be 2.5 kg of $^{239}$Pu in a meter of ABR spent fuel, or from a delayed-neutron perspective, approximately 1 kg of $^{235}$U. This quantity of fissile material is a factor of 2 lower than in the spent fuel processed in the FAST facility. Hence, a stronger interrogating source strength is needed. Increasing the interrogating source strength by a factor of 2 to ~1 × $10^{13}$ n/s would suffice.

The intensity of ~1 to ~3 × $10^{13}$ n/s is right at the limit of what is presently seen as possible for neutron generators using a deuterium and tritium (DT) beam.[8, 9] If a more accurate estimate is desired, a Monte Carlo model would be needed as well as direct interaction with the ABR spent-fuel researchers. The point of the above arguments is that it is reasonable to expect that DT neutron generators can produce enough neutrons to measure the most intense spent fuel expected in the presently envisioned GNEP fuel cycle. Furthermore, it is worth noting that the intensity of the interrogating source was driven by ABR and MOX LEU fuel. The $UO_2$ LEU prevalent in the USA is more than an order of magnitude weaker in neutron intensity. A final relevant point is that a limiting factor in neutron generator development is the difficulty in getting a large beam current into a small target. Spent-fuel measurements do not need the target material to be small. In fact, it would be beneficial if the target surrounded the fuel.

### 4.1.3 Final analysis needs

The main point of the previous section is to show that it is possible to produce a strong enough burst of neutrons that can induce enough fissions in the spent fuel to create a delayed-neutrons signal which can be discerned above the level of the neutrons inherently emitted from the spent fuel. For shipper/receiver, that fact may suffice. However, for some of the other reasons outlined in the introduction, quantifying plutonium is also necessary. In order to do that, it is necessary to determine the isotopics of the major delayed-neutron sources. If the neutron source is below 1 MeV, this generally means it is $^{235}$U, $^{239}$Pu, or $^{241}$Pu. Since a DT-generator-based delayed-neutron diagnostic produced higher energy neutrons (approximately 2 MeV for the $^{252}$Cf shuffler vs. 14 MeV for the DT generator), the delayed neutron from $^{238}$U may be important.

Two paths are envisioned for addressing the need for isotopic data. The first approach is to use gamma and neutron measurements, presently used to quantify burnup and cooling time, to calibrate a delayed-neutron instrument. The second approach is to combine a delayed-neutron instrument with some of the instruments described later in this document. A few details will be given here about the first approach, and the second approach will be addressed with the other instruments. By measuring $^{134}$Cs/$^{137}$Cs, the burnup (exposure) can be quantified. For a given burnup, the neutron intensity is a function of initial enrichment. From the $^{137}$Cs signal, the total number of fissions can be quantified. By combining these neutron and gamma measurements, it should be possible to make a calibration curve giving the mass of plutonium as a function of delayed-neutron intensity.

### 4.1.4 How applicable is this measurement approach?

In the introduction, five motivations for measuring spent nuclear fuel were described. A neutron-generator-based delayed-neutron instrument is likely to be applicable in four of the five circumstances. The one situation where it would probably not work is for the IAEA, which needs portable and low-cost

systems. Given the power needs of such a strong neutron-generator-based system, a spent-fuel delayed-neutron measurement system would not be very portable or inexpensive. The other spent-fuel measurement situations listed in the introduction involved large nuclear facilities engaged in reprocessing, storage, fuel fabrications, or power generation. For power plants, because of the possible low frequency of shipments, one instrument might be shared among several plants.

## 4.2. Differential die-away

### 4.2.1 Concept

A differential die-away technique (DDT) quantifies the fissile material in a sample by measuring the prompt neutrons from induced fissions. The number of neutrons from a given isotope is proportional to the mass of that isotope. In order to interpret the prompt-neutron signal, the isotopic composition of the major actinides must be determined by other means. The neutrons used to induce fissions generally come from a DT neutron generator.

In order to discern the neutrons coming from the fissile material from those coming from the neutron generator, a neutron detector is used that only detects neutrons above 1 eV, and data is only taken after the neutrons from the neutron generator have moderated to below 1 eV in energy.[10] Typically, this means that neutrons are counted between 1 ms and 4 ms after the neutron-generator pulse. Note that the neutrons from induced fissions are born with approximately 2 MeV of energy; thus, they can be distinguished from the less than 1 eV neutrons remaining from the initial neutron-generator pulse during the time interval between 1 ms and 4 ms. A 100-Hz repetition rate is representative for a DDT system.

### 4.2.2 Design options

The primary design concern with a delayed-neutron system was the technical feasibility of producing a strong enough interrogating neutron source. Since the interrogating neutron source for both instruments, in the context of LWR and ABR spent fuel, is a DT neutron generator, and since the DDT system used neutrons more efficiently, it is clear from the previous section (4.1) that a strong enough neutron generator exists. The primary reason the DDT system is more efficient with neutrons is that it detects prompt neutrons instead of delayed neutrons.

However, since the neutrons interrogating the spent fuel during the time interval of interest are thermal (1 ms to 4 ms, as described in section 4.2.1), self-shielding is a concern. When measuring a single rod, self-shielding will not be a concern. If the fission rate per isotope varies noticeably as a function of radius, this effect can be quantified due to the controlled geometry, and a correction factor can be applied. When measuring full assemblies, self-shielding needs to be investigated to determine the relative strength of the induced fission signal among the rods. This will be quantified with Monte Carlo modeling in the near future.

### 4.2.3 Final analysis needs

As in a delayed-neutron system, isotopics are needed to interpret the origin of the detected neutrons. DDT is different from a delayed-neutron system because the interrogating neutrons only fission fissile isotopes. The methods described for determining the isotopics for the delayed-neutron technique apply here as well. Now that the physics of the DDT has been described, it is useful to note how it complements a delayed-neutron system. As discussed earlier, $^{235}$U provides a stronger delayed-neutron signal than $^{239}$Pu by a factor of approximately 2.6. With a DDT system, the $^{239}$Pu provides a stronger signal since the fission cross section is approximately 1.3 times greater than that with $^{235}$U for thermal neutrons. A hybrid delayed-neutron and DDT instrument would provide complementary information.

### 4.2.4 How applicable is this measurement approach?

With respect to the five motivations for measuring spent nuclear fuel, described in the introduction, a DDT system has the same applicability as a delayed-neutron system—applicable to 4 of the 5 motivations. It would be too large and costly a system for use by the IAEA.

## 4.3. Lead slowing-down spectrometer

### 4.3.1 Concept

The lead slowing-down spectrometer (LSDS) is capable of quantifying the amount of $^{235}$U, $^{239}$Pu, and $^{241}$Pu. Physically, the instrument functions by passing the spent fuel through the center of a large cube of lead. An active neutron source (a Linac-driven photoneutron

target) sends out a burst of neutrons (evaporative spectrum) from near the center of the lead cube.[11] These neutrons slow down gradually since each collision with lead reduces the energy of the neutron very little. As these neutrons slow down, the presence of different materials can be determined from the resonances (steep increases in a cross section over a narrow energy range) in their fission cross section. Since the energy of the neutrons is known as a function of time, the neutron flux is known as a function of energy. While slowing down, the neutrons are continually interrogating the item and inducing fission. The neutron flux from induced fission is measured as a function of time by fast neutron detectors embedded in the lead. Given the unique fission cross-sectional data of $^{235}$U, $^{239}$Pu, and $^{241}$Pu, the quantities of each isotope can be obtained by detecting fast neutrons emitted as the neutron burst gradually slows down in energy.

#### 4.3.2 Design options

The major design criteria for a neutron slowing-down spectrometer is creating a neutron pulse that slows down gradually while keeping a narrow spread in energy. High-purity lead is an ideal material for producing this condition since the energy of a neutron is only reduced approximately 1% with each collision and since the purity of the lead keeps the neutron energy distribution narrow. The introduction of hydrogen on a part-per-million level noticeably deteriorates the performance of the LSDS.[11] In this context, a key design concern for measuring spent fuel is the presence of hydrogen in the fuel. In many cases the fuel to be measured may have been in a spent pool recently. It is assumed that a means for drying off the exterior of the fuel would not be too difficult. However, can the instrument work if water has leaked into some of the rods? An important first step in assessing the applicability of an LSDS is to research what is known about the leakage of water into the wide range of spent fuel prevalent today. Based on the results of the leakage research, how well can the LSDS perform over the range of possible water content in spent fuel? Furthermore, would it be acceptable to have an instrument that cannot measure a certain percentage of the spent fuel at a facility? Could special procedures be included to remove the water from inside rods?

Another design activity would be to calculate the intensity of the neutron pulse needed to observe the induced fast fissions above the inherent neutron source in the spent fuel. What would be the cost, size, and maintenance issues for such a Linac? Note that because of the hydrocarbon coolants used in DT neutron generators, the discussions of neutron source strength in the context of a delayed-neutron device are not relevant here.

The part of the $^{239}$Pu fission spectrum of most interest is approximately 0.3 eV because the fission cross section for $^{239}$Pu peaks at that energy. In the context of self-shielding, the significance of the 0.3 eV peak puts the LSDS approximately one order of magnitude higher in energy than a DDT system and at least one order of magnitude lower in energy than a possible delayed-neutron instrument. However, given the ability to quantify any self-shielding present because of the well-known geometry, it is not expected that self-shielding will be an issue for the use of the LSDS with spent fuel.

#### 4.3.3 Final analysis needs

The final analysis needed with the LSDS is simpler than that for the majority of the other techniques described in this article. This is a very attractive aspect of this instrument; the instrument functions by detecting the fast fission neutrons directly for the key isotopes of $^{235}$U, $^{239}$Pu, and $^{241}$Pu. Presumably, a calibration curve can be made that correlates the fast fission neutrons detected at specific energy ranges with the quantities of each of these isotopes. This situation is in contrast with the other techniques described in this article that generally need two or three unique measurements in order to quantify the plutonium content in the spent fuel.

#### 4.3.4 How applicable is this measurement approach?

With respect to the five motivations for measuring spent nuclear fuel, as described in the introduction, an LSDS is likely to have the same applicability as a delayed-neutron or DDT system—applicable to 4 of the 5 motivations. It would be too large, heavy, and costly a system for use by the IAEA. One point that needs to be researched further is the size and cost of the Linac system needed to provide a neutron pulse of sufficient strength for measuring spent fuel.

## 4.4. Neutron resonance absorption

### 4.4.1 Concept

Similar to the LSDS, nuclear-resonance absorption uses the unique resonant structure in the cross sections of the major actinides of interest to quantify the presence of these actinides in the spent fuel. The two techniques differ because the LSDS, by detecting fission neutrons, is only sensitive to the fission cross section of each isotope; in contrast, nuclear-resonance absorption, by detecting changes in the energy spectrum over a narrow energy range, is sensitive to the total cross section of each isotope.

The fundamental concept is that each isotope absorbs neutrons particularly well over a narrow energy range. How well they absorb neutrons can be used to quantify them.[12] The neutrons detected in this technique originate inside the spent fuel. A nuclear-resonance absorption instrument functions by positioning neutron-absorbing foils (gadolinium and cadmium) in conjunction with $^{235}$U fission foils in very close proximity with the spent fuel. The neutron-absorbing foils select the energy range of neutrons that make it to the fission foils. It is the relative change in the energy spectrum leaving the surface of the fuel that is altered by the content of the fuel. This change in the energy spectrum is used to quantify what is inside the spent fuel. For example, the resonance in $^{239}$Pu occurs around 0.3 eV. Samples containing little or no $^{239}$Pu will emit neutrons that have a smooth energy spectrum around 0.3 eV, whereas samples containing significant $^{239}$Pu will have a depression in the neutron energy spectrum at the surface of the spent fuel at an energy of 0.3 eV.

### 4.4.2 Design options

A major advantage of the nuclear-resonance absorption technique is that it uses the inherent neutron emission from spent fuel to interrogate the spent fuel. It is lightweight and inexpensive and operates in the presence of hydrogen.

The key issue with the nuclear-resonance absorption technique is whether the signal obtained can provide high enough quality results. Unlike with the active techniques, there is no knob that can be turned to get better statistics.

The self-shielding concerns would be the same as with the LSDS. Given the well-known geometry of spent fuel, it is not expected that self-shielding will be a major concern. Another issue of uncertainty is extrapolating localized measurements to the entire rod or assembly. The other neutron based technique collect information from the bulk of the spent fuel.

### 4.4.3 Final analysis needs

The final analysis needed with the nuclear-resonance absorption is simple relative to most of the other techniques. This is a very attractive aspect of this instrument, in addition to its being very portable. The instrument functions by detecting the changes in the neutron energy spectrum for the key isotopes of interest: $^{235}$U, $^{238}$U, and $^{239}$Pu. Presumably, a calibration curve can be made that correlates the change in the neutron energy spectrum over a specific energy range with the quantities of each of these isotopes. This situation is in contrast with the other techniques described in this article that generally need two or three unique measurements in order to quantify the plutonium content in the spent fuel.

### 4.4.4 How applicable is this measurement approach?

With respect to the five motivations for measuring spent nuclear fuel, as described in the introduction, a nuclear-resonance absorption technique is applicable to all. It would be portable and low cost. The main issue to be quantified is how accurate it can be.

## 4.5. Passive multiplicity counting

### 4.5.1 Concept

The fundamental physics concept of passive multiplicity counting[13] is that multiple neutrons are generally released when a nucleus fissions. Since these neutrons are released at the same time, they will be detected at nearly the same time. By making time-resolved neutron measurements, neutrons from fission taking place inside a detector can be discerned from background neutrons such as ($\alpha$, n) reactions or fission neutrons occurring outside the detector. Since curium is the dominant source of spontaneous neutrons in spent fuel, the multiplicity distribution will be dominated by curium. However, given that multiplication events populate the higher elements in the distribution, multiplicity counting is sensitive to multiplication in the sample; hence, the fissile contents can be quantified. The ratio of curium

to plutonium or isotopic information is needed to determine the plutonium mass.

### 4.5.2 Design options

Until recently, multiplicity counting of spent fuel would not have been considered possible since the count rates were too elevated to be handled by the multiplicity electronics of a reasonably efficient counter. The introduction of list-mode data acquisition and fast, inexpensive computation power has allowed such high count-rate applications by fundamentally changing the nature of the data processing. Previously, multiplicity analysis was performed by combining all the signals from all the detector tubes together so that the electronics could determine the number of neutrons arriving within a given time window or gate; this process limited the count rate that could be handled. Now, with improvements in the amount of data that can be stored, as well as improvements in computational speed, it is possible to store data from the individual tubes of the overall detector and perform the analysis after the measurement. This allows a new approach to handling high-count-rate applications: simply make many small, independent subsections of the detector and analyze the combined data from all the subsections after the measurement. For the design, this means many small detector tubes will enable high-count-rate application.

### 4.5.3 Final analysis needs

The data collected with a passive multiplicity counter using list-mode data acquisition can be as simple as a list of the instants in time when the pulses were detected. To get the mass of plutonium from this, it will be necessary to get some isotopic information, as was described in the delayed-neutron section. The isotopic data is needed to quantify the amount of curium, the multiplication in the spent fuel, and the mass of plutonium.

Because of the dominance of curium as the neutron source in spent fuel, one analysis option is to quantify curium and then use the ratio of curium to plutonium to quantify the plutonium in spent fuel. If the curium to plutonium ratio is to be used to quantify plutonium, it is important to understand how this ratio varies in a reprocessing plant. An example where a difference in this ratio is observed is between the accountability tank with respect to the ratio in the hulls. A calculation was done to better understand the variation in this ratio; the variation in the ratio of curium to plutonium as a function of radius is depicted In Fig. 2. These data are the results of an MCNPX/Cinder calculations. A rod (PWR, 4.1% $^{235}$U) was divided up into 9 radial sections and 5 axial sections. The results below were from the central section. The average burnup of the rod was 39.4 GWday/MTHM. The radial location of the data points illustrated in Fig. 2 is at the radial midpoint of the respective volume elements. What is observed is a 64% decrease in the plutonium to curium ratio from the central voxel to the exterior voxel with a pronounced reduction over the exterior of approximately 0.1 mm. Quantifying this effect will assist in using the ratio of curium to plutonium to quantify plutonium.

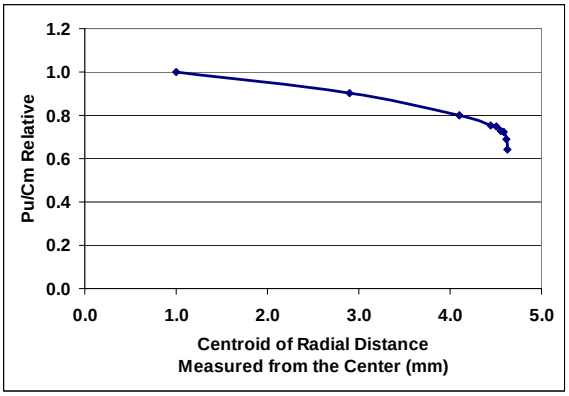


Fig. 2. A variation in the ratio of plutonium to curium in one PWR rod (39.4 GWday/MTHM burnup) with an initial enrichment of 4.1% $^{235}$U is depicted. The rod was divided up into 9 radial sections and 5 axial sections. The results are from the central section.

### 4.5.4 How applicable is this measurement approach?

With respect to the five motivations for measuring spent nuclear fuel, as described in the introduction, a passive neutron multiplicity system would probably work in all contexts. It could be small enough and low enough in cost for the IAEA. Furthermore, in the context of shipper/receiver differences, it may be acceptable to simply measure the multiplicity distribution of an assembly as it passes through a ring detector at one site and then reproduce that measurement at the receiving site to assure that the same object leaving one facility arrives at the other. It would be very difficult to take material from a fuel assembly and add something in a way that would reproduce the multiplicity distribution, especially considering the effect of multiplication on the distribution. Note that multiplicity analysis can be incorporated into a

DDT or delayed-neutron instrument at little additional cost.

## 4.6. Passive neutron albedo reactivity

### 4.6.1 Concept

The technique functions by using the intrinsic neutron emission of the fuel (primarily from the spontaneous fission of curium) to self-interrogate the fissile material in the fuel itself.[14] Two separate measurements of the spent fuel are made, and the ratios of the count rates obtained are analyzed. The primary difference between the two measurements is the neutron energy spectrum in the spent fuel. By varying the material around the spent fuel, a high and a low neutron-energy-measurement condition can be produced. The neutron detectors can be used to detect total neutrons (singles) and/or doubles and/or triples; it is expected that doubles will produce the best result in the high count-rate regime.[15]

One approach to producing these two energy conditions involves measuring the spent fuel in air with a thin layer of cadmium surrounding it (located between the spent fuel and the moderating walls of the detector). The other approach involves no cadmium at all. The cadmium effectively eliminates all neutrons below 1 eV from reflecting from the detector walls back into the fuel. Hence, in the measurement made with no cadmium present, the fuel is interrogated by all the neutrons reflected back to the fuel. In contrast, when cadmium is present, the fuel is interrogated by only those reflected neutrons with energies above 1 eV. Since the only difference in the two interrogating spectrums is the presence of neutrons below 1 eV, how far the ratio deviates from unity indicates the impact of the neutrons below 1 eV.

An example of some simulated data is reproduced in Fig. 3 below. The Cadmium Ratio is the count rate when no cadmium is lining the inside of the detector chamber divided by the count rate when there is cadmium present. For the data in Fig. 3, plutonium was removed from an oxide mix containing neptunium, plutonium, americium and curium in the proportions expected in LWR spent fuel.

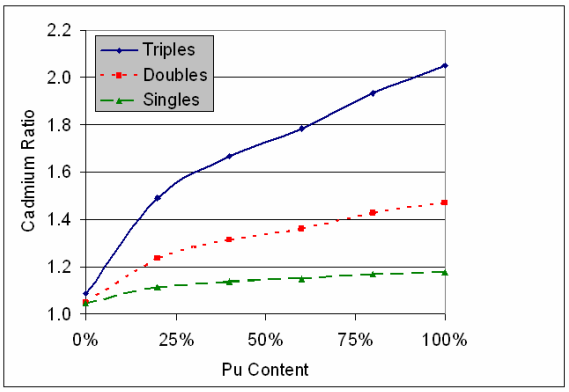


FIG. 3. The modelled Cadmium-Ratio results for the UREX$^{+1a}$ product material..

### 4.6.2 Design options

A depiction of an experiment performed to test the PNAR concept is depicted in Fig. 4.[14, 15] A likely implementation of the PNAR concept with spent fuel would have two cylindrical rings; the rings would be identical except one ring would be wrapped in cadmium and the other would not. The two rings would be separated by approximately a meter, and the spent fuel would move simultaneously through both rings. There would be fewer $^{3}$He tubes than depicted in Fig. 4, and the walls of the detector would be much thinner.

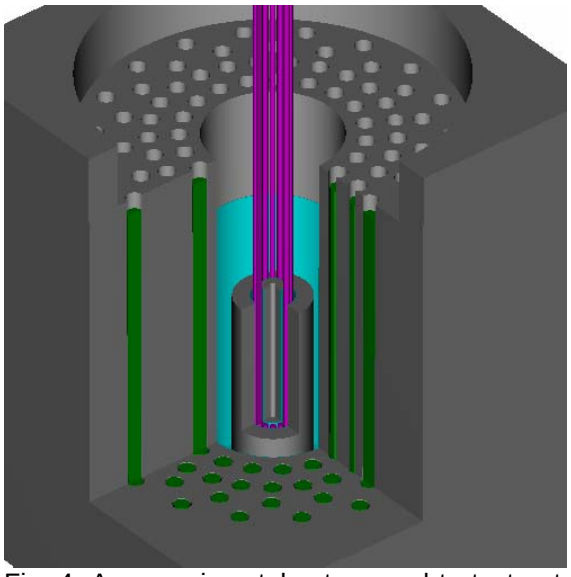

Fig. 4. An experimental setup used to test out the PNAR concept with fresh LEU fuel and a Cf source. Green cylinders are $^{3}$He, pink rods are LEU fresh fuel, grey is polyethylene and light blue is cadmium.

Since PNAR is effectively interrogating with thermal neutrons, self-shielding is a concern in the same way that it is with DDT. As with the DDT, self-shielding needs to be quantified for an assembly to see if the well-known geometry

of spent fuel can be leveraged to quantify self-shielding and correct for it.
Since the induced fission rate at thermal energies is 1.3 times stronger for $^{239}$Pu than for $^{235}$U, combining this measurement with a delayed-neutron measurement is beneficial and complementary from a hardware standpoint. An interesting combination that would build on the split-ring design discussed above would be to measure delayed neutrons with each of these rings as well.

An interesting ramification of using the neutrons from the spent fuel to interrogate the spent fuel is that the statistics get better as the inherent neutron source in the spent fuel becomes more intense.

#### 4.6.3 Final analysis needs

The isotopics concentrations of the fissile isotopes must be determined in order to obtain the plutonium mass. The design of this detector is essentially identical to one designed to do passive multiplicity measurements.

#### 4.6.4 How applicable is this measurement approach?

With respect to the five motivations for measuring spent nuclear fuel, as described in the introduction, a PNAR system would likely work in all contexts. It could be small enough and low enough in cost for the IAEA; it would effectively be the same as two passive multiplicity counter units. Furthermore, in the context of shipper/receiver differences, it would provide the multiplicity distribution of an assembly as well as the cadmium ratio value. It would be very difficult to take material from a fuel assembly and add something in a way that would reproduce the multiplicity distribution, especially considering the effect of multiplication on the distribution as well as the cadmium ratio.

### 4.7. X-ray fluorescence

#### 4.7.1 Concept

Significant quantities of x-rays are emitted from spent fuel. These x-rays are stimulated by the radiation emanating from the spent fuel, primarily $^{137}$Cs for the cooling times of interest to spent fuel measurement. The elemental ratio of plutonium to uranium in the edge layer of the spent fuel can be determined by measuring these x-rays. The ratio of plutonium to uranium in spent fuel varies across a given rod in a predictable way so that the measured ratio on the edge of the rod can be used to determine the average ratio over the rod. Then, if the total uranium in the rod is determined using some of the techniques listed in the delayed-neutron-technique section, the total plutonium can be determined from the product of the average plutonium-to-uranium ratio and the total uranium.

#### 4.7.2 Design options

Among the seven instruments described in this article, this is the only technique that does not detect neutrons. The actual measurement of x-rays from a spent-fuel assembly is straightforward and can be conducted in air or water. Collimation can lower the count rate to whatever level is desirable.

#### 4.7.3 Final analysis needs

In the concept section, it was stated that “the ratio of plutonium to uranium in spent fuel varies across a given rod in a predictable way.” This statement describes the need to extending the x-ray measurement made on the surface of the fuel to the bulk of the fuel. This extension from the surface to the bulk needs to be quantified for a range of burnups, and the uncertainty in this extension must be quantified. Also, the uncertainty of making a localized measurement on the edge of an assembly and extending this to the full assembly must be quantified.

#### 4.7.4 How applicable is this measurement approach?

The relative mass of plutonium to uranium would assist in the analysis of all the other techniques as well as possible working to quantify fissile mass as an independent concept. All of the other techniques involve separating the properties of plutonium and uranium in some way. An independent technique directly measuring this difference would be of assistance universally.

With respect to the five motivations for measuring spent nuclear fuel, as described in the introduction, an x-ray–based system would likely work in all contexts. It could be small enough and low enough in cost for the IAEA.

## 5. Acknowledgements

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## 6. Summary

Five motivations for measuring the fissile or plutonium content in spent fuel were described. The following is a brief summary of them: an input accountability tank for pyrochemical reprocessing, independent verification for the IAEA, shipper/receiver differences, spent LWR fuel very similar to fresh ABR fuel, and increasing quantity of "non-self-protecting" spent fuel. The neutron intensities expected from LWR and ABR spent fuels were quantified to assist in assessing the feasibility of several of the measurement techniques. The range in intensity for 5-year-cooled fuel is expected to range from ~3 × $10^7$ n/s to ~3 × $10^9$ n/s per meter for full burnup LWR and metal ABR fuel. The upper limit has a high degree of uncertainty given the unknown design of ABR fuel; note that the equilibrium value obtained from recycling actinides in the ABR fuel multiple times was used for the intensity of spent ABR fuel.

The physics concepts of seven measurement techniques were described with emphasis given to the need for additional information from other instruments to quantify the plutonium mass. Also, in some cases it was pointed out how the instrument might serve in a unique way in the shipper/receiver context. An effort was made to indicate how several of the techniques could work in concert to improve the results of any one technique. Four of the techniques (neutron resonance absorption, passive multiplicity counting, PNAR, and x-ray fluorescence) were identified as being lightweight and inexpensive enough to be of interest to the IAEA. The next step is to start to test these concepts with modeling and through measurements in the context of spent fuel to determine the most accurate technique or combination of techniques.

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# IMAGE PROCESSING TECHNIQUES FOR VERIFICATION OF LONG-COOLED SPENT FUEL AND PARTIAL DEFECT DETECTION USING THE DIGITAL CHERENKOV VIEWING DEVICE

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***Abstract:***

*The Digital Cherenkov Viewing Device (DCVD) is now in routine use for long-cooled spent fuel verification by the International Atomic Energy Agency (IAEA). Image processing has been implemented to assist inspectors in the use of the DCVD. Additionally, the potential to apply image-processing techniques on acquired images of spent fuel results in higher understanding of the images and its features. We have carried out studies using the DCVD to find missing or substituted pins in spent fuel assemblies. Used imaging techniques are described herein. The initial results gathered from field tests indicate that the DCVD could be used as a partial defects tester and not only as a gross defects tester for which it is approved.*

*Added features to the instrument are 1) an alignment aid to ensure that the instrument is properly aligned, 2) a programme that follows the fuel of interest even if the detector is moving and 3) a feature that identifies each fuel position for the inspector. Future work envisages the use of library images to correctly identify the fuel type.*

**Keywords:** image-processing; spent fuel verification; partial defect test; gross defect test

## 1. Introduction

The development of the Digital Cherenkov Viewing Device (DCVD) is a joint program between the Swedish and Canadian Safeguards Support programs that started over ten years ago. The DCVD has been in use for some time by the International Atomic Energy Agency (IAEA) and has up until now primary been utilized for verification of long-cooled spent fuel. The program is now trying to improve the instrument and add new functionality to better assist the inspectors in their work, and give the instrument new usage areas. We foresee great potential in applying image-processing techniques on acquired digital Cherenkov images to achieve these objectives [1]. The image-processing techniques described herein indicate that the DCVD could be used as a partial defects tester and not only as a gross defect tester for which it is already approved.

## 2. The Digital Cherenkov Viewing Device (DCVD)

The DCVD images the ultraviolet Cherenkov light (wavelength 280-340 nm) that is generated in the water by gamma radiation emitted by fission products in the spent fuel. The instrument is comprised of: a digital ultraviolet-sensitive camera from Andor Technology (Ireland); a railing-bracket to position and align the camera properly over the fuel assemblies in the fuel bay; a customized computer with a touch-screen running a software program to control the camera and display settings, and to acquire, save, and review acquired digital Cherenkov images of the fuel [2].

**Figure 1: The DCVD in use**

### 2.1. Typical DCVD fuel images

Over the years the DCVD has acquired a number of images of spent nuclear fuel.

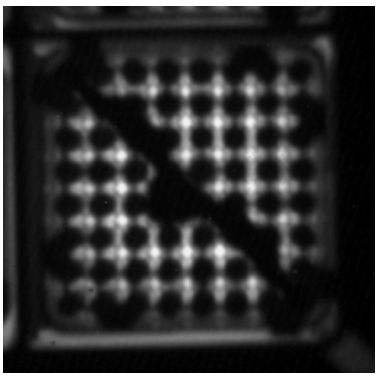

**Figure 2: BWR Cherenkov image**

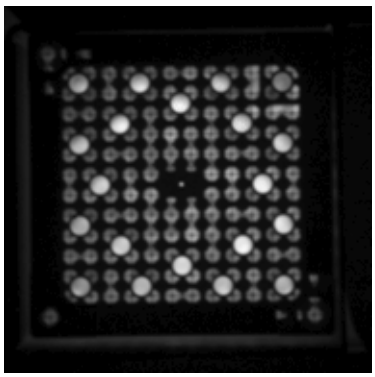

**Figure 3: PWR Cherenkov image**

A significant Cherenkov characteristic of spent fuel is the collimation effect from light emitted in the water columns between the fuel pins. In the DCVD images (figure 2 and 3) of boiling water reactor (BWR) and pressurized water reactor (PWR) spent fuel, the bright water columns and the dark fuel pins are easily identified. The light intensity from a fuel varies significantly with the alignment of the camera relative to the fuel. If the camera is positioned to the left of the fuel assembly, then the light contribution from the left part of the fuel assembly is greater than the light contribution from the right part of the fuel assembly et cetera.

## 2.2. Instrument alignment aid

The instrument alignment aid is a method used to help align the DCVD vertically and horizontally when acquiring images of the spent nuclear fuel for verification. Ideally, when aligned over a fuel assembly the emitted Cherenkov light is uniformly distributed from the water columns between the fuel pins over the fuel assembly, i.e. the light emitted from the left half of the fuel assembly equals the light emitted from the right half of the fuel, and the light emitted from the top half of the fuel equals the light emitted from the bottom half of the fuel.

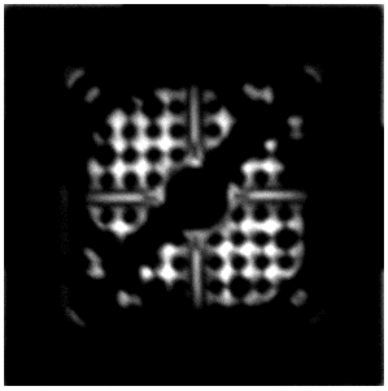

**Figure 4: Properly aligned over the fuel**

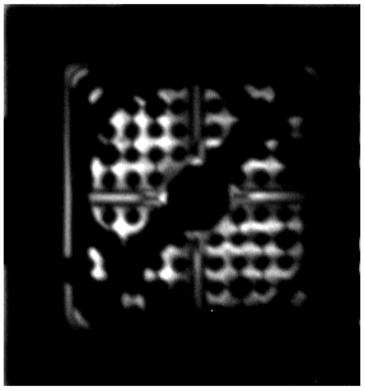

**Figure 5: Positioned to the left of the fuel**

The figures above visualize horizontal off-alignment: In figure 4 the instrument is properly aligned over the fuel (notice that the vertical part of the water-cross is illuminated) and in figure 5 the instrument is positioned to the left of the fuel.

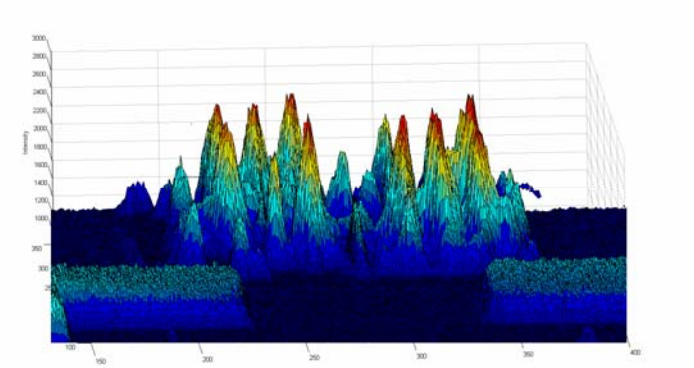

**Figure 6: 3D-image of the light distribution from fuel where the instrument is properly aligned**

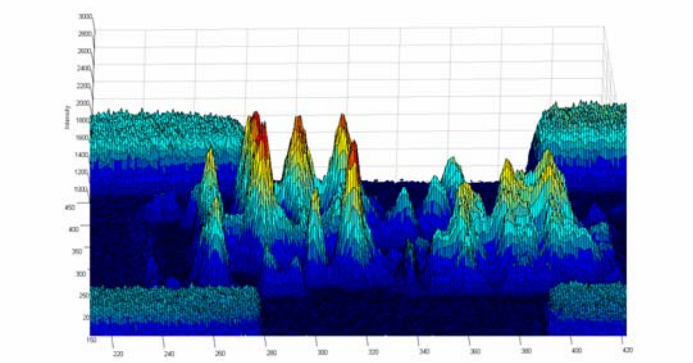

**Figure 7: 3D-image of the light distribution from fuel where the instrument is positioned to the left of the fuel**

By analyzing the distribution of the emitted Cherenkov light from a fuel assembly in the acquired images, it is possible to indicate if the instrument must be moved vertically and/or horizontally to get the instrument aligned. To assist the inspectors we assess that the method must: display alignment information to the user to indicate to the user the direction and magnitude to move the instrument; update the alignment information frequently; be reliable; and be possible to enable/disable from the instrument ‘s user interface. Our method calculates the centre of the intensity in the region of interest (the fuel to be verified) with frequent intervals, resulting in the alignment deviation in the horizontal direction and alignment deviation in the vertical direction. To calculate the centre of intensity we use a filtering algorithm that enhances and separates the light from each water column in the fuel. A filtered image gives us the possibility to determine the size of the individual light sources within the fuel assembly. By analyzing the sizes of the individual light sources in the region of interest, we can decide the deviation from alignment. The method must furthermore be able to handle non-ideal conditions, such as hot/cool neighbouring fuel, asymmetry aspects, and water turbulence.

If the fuel to be verified has neighbours with different burn-up and cooling time, the Cherenkov light contribution from those neighbours within the region of interest will typically not be the same. For example, if the fuel to be verified has a hot neighbour with short cooling-time to the left and a cool neighbour with long cooling-time to the right, the light intensity contribution within the region of interest from the hot neighbour on the left will be greater than the light intensity contribution from the cool neighbour.

Typically, a fuel assembly is not geometrically symmetrical in both horizontal and vertical direction, e.g. many fuel types have a handle that covers the fuel assembly asymmetrically. Furthermore, there are other reasons for an asymmetrical light distribution over the fuel assembly, e.g. missing fuel pins or substituted fuel pins in the fuel assembly will create an asymmetrical light distribution.

Water turbulence in the fuel pond can create an optical effect where the emitted Cherenkov light from a fuel assembly is refracted differently for different regions of a fuel assembly when the light emerges from the water into the air. As a result, the light intensity from the different regions of a fuel assembly will vary over time.

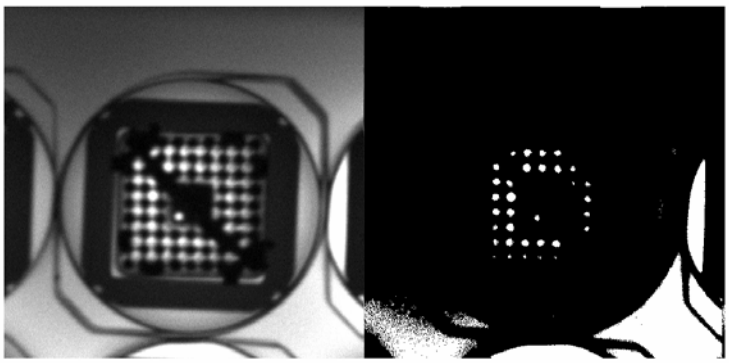

**Figure 8: Original Cherenkov image to the left and filtered image to the right.**

To meet the usability requirements and to minimize the effects from non-ideal conditions the method is executed in the following steps:

1. Filter the image to enhance and separate the light from each water column in the fuel.
2. Divide the region of interest in equally sized left and right regions.
3. Calculate the size of all the light sources in the left and right regions respectively.
4. Find the median size of a light source in the left and right regions respectively.
5. Compare the median size of the light sources in the left and right regions.
6. Calculate deviation from perfect alignment in the horizontal direction based on the comparison in step 5.

For the calculation of the deviation from perfect alignment in the vertical direction, divide the region of interest into equally sized top and bottom regions and repeat step 3 to 6. One challenge with implementing this method is to find a stable filtering algorithm, i.e. a filtering algorithm that really separates the light sources from each other for all fuel types and all light conditions. Another challenge is to perform the steps in the method fast enough, i.e. the graphical alignment aid feedback to the user must be delivered within a very short period of time after the image acquisition. We have already made some initial tests in the field with a dynamic filtering algorithm that shows promising results. The remaining parts of the method have been tested with positive results on previously acquired Cherenkov images.

## 2.3. Automatic target recognition

For every image that is acquired by the DCVD, a set of computing intensive algorithms operates on the area in the image covering the fuel assembly to be verified. This gives the optimal image with respect to contrast and brightness. For the software to know what area to use in the calculations, the user manually defines a region of interest. When the instrument is moved the region of interest must be redefined, i.e. moved. This is annoying for the user and can be very time-consuming when aligning the instrument over a fuel assembly. Automatic target recognition is a method to recognize a specific image area (region of interest) in the images acquired by the camera. Thus, if the user once defines the region of interest and then moves the instrument in some direction, the region of interest should automatically follow the same image area and no further manual interaction is needed. To assist the inspectors we assess that the method must: automatically move and display the region of interest (follow the fuel) in the user interface with frequent intervals when the instrument is moved; work under normal conditions, i.e. some water turbulence, light reflections, etc.; be reliable; and be possible to enable/disable from the user interface. This can be achieved using a phase correlation method. Phase correlation is a computationally efficient frequency domain approach to determine the relative translative movement between two sequentially acquired images.

The standard phase correlation method can be described with the following steps:

Given two input images $i_a$ and $i_b$, where $i_b$ is a translation of $i_a$. Apply a window function (e.g. the Gauss band window) on both images to reduce edge effects. Then, calculate the discrete 2-dimensional Fourier transform of both images.

$$I_a = \Im\{i_a\}, I_b = \Im\{i_b\}$$

Take the complex conjugate of $I_b$. Multiply the Fourier transforms together element-wise. Normalize the product element-wise to get *N.*

$$N = \frac{I_a I_b^*}{\left| I_a I_b^* \right|}$$

Inverse transform *N* to get the phase correlation, *PC.*

$$PC = \Im^{-1}\{N\}$$

And then determine the peak in the inverse transform.

$$(\Delta x, \Delta y) = argmax_{\Delta x \Delta y}\{PC\}$$

The $\Delta x$ and $\Delta y$ then gives the translation of $i_a$ in the horizontal and vertical direction. The proof of the phase correlation method is based on the Fourier shift theorem.

The automatic target recognition method must be able to handle non-ideal conditions, such as the collimation effect, water turbulence, low light intensity, and camera movement together with long exposure times.

When moving the instrument over a fuel, the light distribution from the fuel varies with the collimation from the emitted light from the fuel. For example, when the instrument is positioned to the left of the centre of the fuel, more light is emitted from the left side of the fuel, and when the instrument is positioned to the right of the centre of the fuel, more light is emitted from the right side of the fuel. This effect will decrease the peak in the phase correlation, because the objects in the image that are used for correlation have changed between two sequential translated images.

As stated in earlier, water turbulence in the fuel pond can create an optical effect where the emitted Cherenkov light from a fuel assembly is refracted differently for different regions of a fuel assembly when the light emerges from the water into the air. As a result, the light intensity from the different regions of a fuel assembly will vary over time. In the same way, as for the collimation effect, this will decrease the peak in the phase correlation.

For fuel with low burn-up time and/or long cooling time, the dynamic range in the acquired images is very low, i.e. the difference between dark areas and bright areas in the image is minimal. The smaller the dynamic range an image has, the lower and less well-defined the peak in the phase correlation.

When the instrument uses relatively long exposures to acquire images, the correlation between two sequential translated images can be significantly decreased. The reason for this is that when the instrument is moved while collecting light, the acquired images gets blurred. Moreover, the negative effects from the collimation and water turbulence are amplified when using long exposure times, see figure 9 and 10.

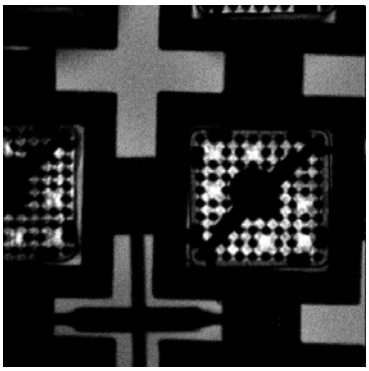

**Figure 9: Sharp Cherenkov image.**

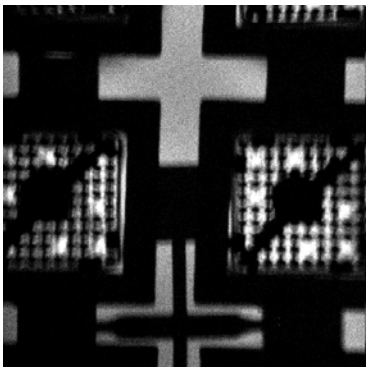

**Figure 10: Blurry Cherenkov image as an effect of instrument movement while collecting light.**

To meet the usability requirements and handle the non-ideal conditions we suggest that the method is executed in the following steps:

1. Select the total image area $i_a$ from the original image.
2. Acquire a new image and select the total image area as $i_b$ from this the second image.
3. Run the phase correlation method and find an isolated and well-defined peak in the resulting inverse transform.
4. Decide which direction the instrument is moved in and move the region of interest $\Delta x$ and $\Delta y$ representing the translation of $i_a$. Update the original image with the latest acquired image.
5. Acquire a new image (the second image) and run the phase correlation method for the direction (using either the vertical or horizontal phase correlation method) that was decided in step 4 and find the peak in the resulting inverse transform.
6. If the peak value is isolated and well-defined, move the region of interest $\Delta x$ and $\Delta y$ representing the translation of $i_a$. Update the original image with the latest acquired image and go to step 5. If an isolated and well-defined peak cannot be found in the inverse transform go to step 1.

One challenge with this method is that it may be too slow due to the amount of data that has to be processed. If the time to complete the phase correlation method takes more time than the currently selected instrument exposure time, e.g. 250 ms, then the resulting translation of the region of interest will be incorrect, i.e. not matching the translation in the currently acquired image. However, initial results from field tests show very positive results. We have been able to meet the requirements set up for the method. On the DCVD the automatic target recognition runs in less than 50 ms using the whole image as input to the phase correlation.

## 2.4. Recognition of missing fuel pins

Recognition of missing fuel pins is a method that finds fuel pins and missing fuel pins in an acquired image of a fuel assembly of known type. A conceptual implementation [3] of the method has already been implemented in and tested in the field on a limited number of fuel assembly types (BWR8, BWR10, PWR15, and PWR17) and the results have been positive under ideal conditions. There are several areas that must be improved for it to be part of the DCVD software. To assist the inspectors the method must: find missing or substituted pins in an image within a reasonable amount of time; highlight missing pins in the user interface; display intensity values for the missing pins in the user interface; be reliable; be possible to enable/disable from the user interface. Moreover, the method must be able to handle non-ideal conditions, such as hidden fuel pins, instrument alignment, low light intensity, water turbulence, and hot/cool neighbour fuel.

For all fuel types a certain number of fuel pins are hidden below a lifting handle or top plate (a physical construction that stops Cherenkov light to be emitted from that region). The hidden fuel pins can never be

detected, and hence the method can never detect all fuel pins (or missing pins) in a fuel assembly. Here is a list of the percentage of hidden fuel pins for different fuel types: BWR 8x8, 14%; 6x6 MOX, 11%; GE6, 19%; GE8, 17%; SVEA64, 28%; SVEA96, 58%; SVEA100, 26%; Atrium 9, 14%; Atrium 10B, 15%; GE12S, 20%; PWR 15x15, 51%; PWR 17x17, 44%; WWER, 52%.

For the method to work and give reliable results, the image that is processed must be acquired with the instrument properly aligned over the fuel assembly. If the instrument is not properly aligned, the light emitted from different fuel pins within the fuel assembly will vary which makes it harder to identify and separate fuel pins and water columns respectively.

For fuel with low burn-up and/or long cooling time, the dynamic range in the acquired images is very low, i.e. the difference between dark areas and bright areas in the image is minimal. The smaller the dynamic range an image has, the harder it is to detect and separate fuel pins from water columns within the fuel assembly.

As described earlier, water turbulence can induce an optical effect that creates blurry fuel images and varying light intensities from different regions of a fuel assembly. If the recognition of fuel pins method is applied to a blurry image the result will most likely not be reliable

If the fuel to be verified has neighbours with different burn-up and cooling time, the Cherenkov light contribution from those neighbours within the region of interest will typically not be the same. If the recognition of fuel pins method is applied to an image where this effect is present, the result will most likely not be reliable

The conceptual implementation is executed in the following steps:

1. Low pass filter the image to remove noise and irregularities.

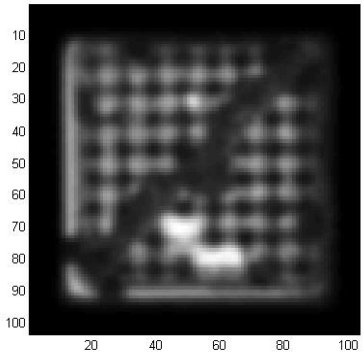


**Figure 11: Low pass filtered Cherenkov image with three missing fuel pins.**

2. Extract orientation and phase information.
3. Filter with a symmetry filter and look for circular objects.
4. Fourier transform the image and extrapolate found positions.
5. Find and label the different objects and their centre position.

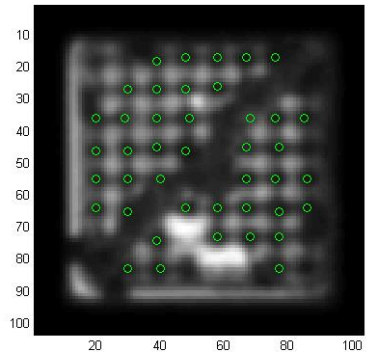


**Figure 12: Found circular objects in the Cherenkov image.**

6. Calculate Euclidian distance between all found objects.
7. Decide smallest distance and main direction.
8. Map input points to grid points.
9. Compare result with a template of the specific fuel type.
10. Find correlation between map and template.
11. Find maximum in the correlation surface to determine relative distance between the map and the template.
12. Calculate image coordinates with help of the grid.
13. Calculate probability for a fuel pin with help of the normal distribution function.
14. Mark fuel pin positions having low probability.

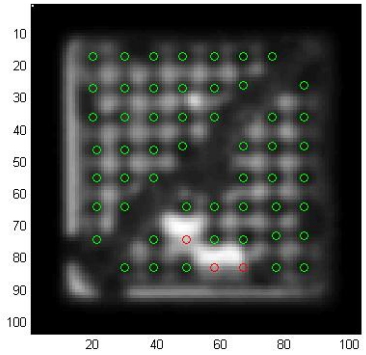


**Figure 13: Cherenkov image where the three missing pins are marked with red circles by the DCVD software.**

Initial tests with the conceptual implementation show promising results under ideal to normal conditions. The method must be refined to meet usability requirements and to handle non-ideal conditions. Moreover, it must be expanded with support for more fuel types. We believe that the recognition of missing pins method opens the door for new usage areas for the DCVD. If the DCVD can indicate to the user which fuel pins that likely are missing or substituted in a fuel assembly, then it could serve as a partial defect tester for the IAEA.

## 3. Conclusions

The three image processing methods described herein will help the inspectors in their work. We foresee great potential in applying image processing techniques on acquired Cherenkov images. We believe that this can broaden the DCVD usage from a gross defects tester to a partial defects tester. The future work will be concentrated around refining the three methods described herein so that they can be part of the DCVD software to be released to the IAEA by the end of the year 2007.

## 4. Acknowledgements

The authors would like to thank the Ringhals Group, owner of the Ringhals nuclear power station and Swedish Nuclear Fuel and Waste Management Group (SKB), owners of the Swedish Central Interim Storage for Spent Fuel (CLAB) for making this study possible. The Swedish Support Program and the Canadian Safeguards Support Program funded this report and the work related to it as part of their support programs for the International Atomic Energy Agency.

## 5. References

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# Cryogenic TES Microcalorimeter Detectors: Safeguard Applications

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## Abstract

A cryogenic transition-edge-sensor (TES) microcalorimeter accurately measures the energy deposited from a single photon or charged particle incident on an absorber maintained at 0.1 K. Resolutions of 25 eV FWHM for 100 keV photons and 2.5 keV FWHM for 5 MeV alpha particles have been obtained. For these energies there is a large improvement in resolution with these types of cryogenic detectors compared to the current Ge-, 500 eV FWHM, and surface-barrier detectors, 12 keV FWHM. The improved resolution has safeguard applications for both destructive (DA) and nondestructive assay (NDA). The improved resolution for alpha spectroscopy allows the major alpha peaks of Pu-239 (5.156 MeV) and Pu-240 (5.168 MeV) to be separated allowing this important isotopic ratio to be determined without the need for mass spectrometry measurements. Gamma-ray applications include improved measurements of the complex gamma and X-ray spectra of special nuclear material in the 100 keV region. Differences in the design of alpha- and gamma-ray detectors will be discussed. This work will also discuss recent cryogenic microcalorimeter alpha (DA) and gamma-ray spectrometry (NDA) measurements on Pu and other nuclear materials carried out by a team from Los Alamos National Laboratory and the National Institute of Standards and Technology/Boulder.

## Introduction

Recent advances in cryogenic microcalorimetry technology have made possible significant improvements in resolution of alpha particle and gamma-ray spectra compared with that available with current semiconductor detection technology. One type of detector that is under development, the transition-edge sensor (TES), uses the change in resistance between the superconducting, zero resistance and normal resistance at very low temperatures, near $0.1^{o}$ K. An absorber, typically Sn, is bonded to the sensor. The energy of an individual alpha particle or photon interacting with the absorber is transformed into heat that momentarily raises the temperature of the sensor and increases its resistance. The absorber must have a low heat capacity so that the interaction of a single photon or charged particle leads to a measurable increase in temperature. The magnitude of the temperature pulse is proportional to E/C where E is the radiation energy and C the heat capacity of the absorber. A resultant current pulse is measured by a SQUID (Superconducting Quantum Interference Detector). This type of system is capable of very high-resolution energy measurements useful for safeguard applications [1].

## Alpha Spectroscopy

Alpha particles emitted by materials important in nuclear material safeguards range in energy from 4 - 6 MeV. Alpha particle spectroscopy on electrodeposited samples using Si semiconductor detectors is routinely used for isotopic analysis. The best resolution for the semiconductor detectors is about 12 keV FWHM (full-width-half-maximum). Some alpha particle energies for different Pu isotopes are close enough so that they are difficult to resolve. One such peak pair is the primary alpha decay branches for Pu-239 and Pu-240. The alpha peaks for Pu-239, 5156.7 keV (Br= 73%), and Pu-240, 5168.2 keV, (Br=73%) are separated by about 12 keV. Another peak pair is Pu-238, 5499.2 keV, (Br=71%) and Am-241, 5485.6 KeV (Br=85%) In both these examples improved resolution from TES detectors can lead to improved estimates of Pu isotopic composition.

Recently the first high-resolution alpha spectroscopy measurements were made with a TES microcalorimeter. Thin samples of polonium and plutonium were prepared and counted with a 250 μM thick, 2.9 $mm^2$ Sn absorber that was bonded to a Mo/Cu TES bilayer, cooled to less than 0.1 K. An alpha spectrum of the 5304.4 keV Po-210 peak that was measured with the TES spectrometer is shown in Figure 1. From this data the detector resolution can be calculated by fitting the higher-energy symmetric side of the peak to a Gaussian form. From this calculation the detector resolution is calculated to be 2.4 keV FWHM, about 4 to 5 times better than that achieved with Si semiconductor detectors.

The low-energy tailing in the peak is probably due to energy straggling due to finite sample thickness. The capability to prepare very thin samples with minimum energy straggling is important in order to exploit the superior TES resolution for alpha spectroscopy. Unlike Si detectors there is no dead layer on the surface of a microcalorimeter to degrade the resolution.

The alpha spectrum for an electrodeposited plutonium sample with 12% Pu-240 is shown in Figure 2. The major high-energy alpha peaks at Pu-240, 5168 keV and Pu-239, 5157 keV are clearly resolved. Other locations of weaker alpha transitions are also indicated in this figure. A normalized spectrum for the same sample measured by a high resolution Si detector is also shown. Low energy tailing of alpha peaks is evident in this spectrum.

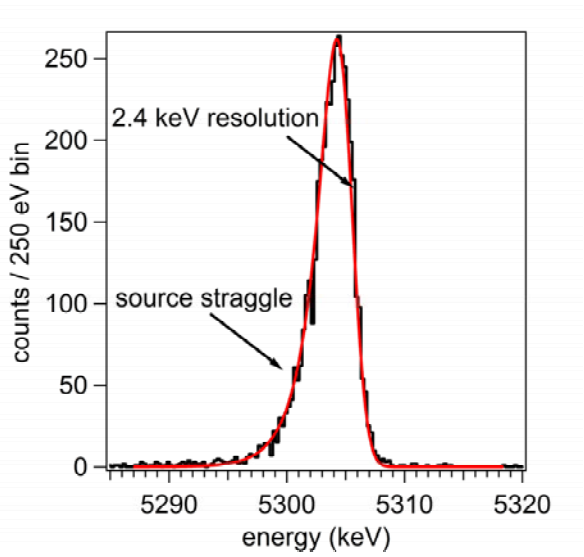


Figure 1. Alpha spectrum of Po-210 measured by a cryogenic TES microcalorimeter.

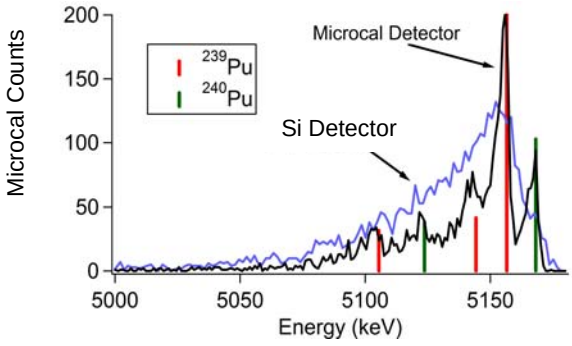


Figure 2. Alpha spectrum of Pu-239/Pu-240 measured by a TES cryogenic microcalorimeter and Si detector.

## Gamma-Ray Spectroscopy

In the late 1990s TES microcalorimeters were developed by NIST for measurement of X-Rays with energies of less than 15 keV for use in

material science applications [2]. More recently TES detectors have been developed that have been used to measure gamma-rays and X-rays from uranium and plutonium at higher energies, greater than 100 keV [3-5]. Resolutions as small as 25 eV FWHM at 103 keV have been obtained for a single absorber (pixel) [5-6]. The gamma-ray spectrum of a 0.4 g Pu standard as measured by a single pixel TES detector is shown in Figure 3. The very high resolution of the TES detector is evident when compared to the spectrum measured with an hpGE detector measurement of the same standard also shown in Figure 3. Also evident in this figure are the poor counting statistics of the Pu isotopic peaks in the microcalorimeter spectrum. About 200 pixels of the area used to collect the TES Pu spectrum is equal to the hpGE detector area. Thus a single pixel, typically with an area of 1 $mm^2$ covers a small solid angle that would require excessively long count times to obtain reasonable counting statistics. The individual pixel area or volume is small to minimize heat capacity and thermodynamic fluctuations to obtain the highest resolution for the spectrometer. For this reason an array of absorbers is required to increase the count rate. Gamma-ray spectra have been measured with arrays as large as 14 pixels with an over all energy resolution of 47 eV at 103 keV, a resolution equivalent to that of the single pixel spectrum shown in Figure 3 [6].

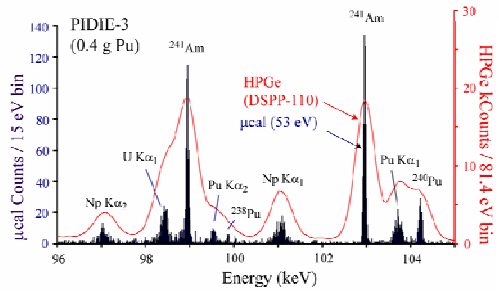


Figure 3. TES cryogenic microcalorimeter spectrum of plutonium sample overlaid with an hpGE planar detector spectrum.

## Discussion

Measurement of isotopic compositions is important in safeguards assays. Gamma-ray measurements are routinely performed with Ge detectors to obtain uranium and plutonium isotopic compositions. The measurements are used to convert the results of thermal power or neutron measurements into total grams of SNM. For calorimetry the conversion factor is $P_{eff}$, the effective specific power ($P_{eff}= \Sigma R_i P_i$, where $R_i$ is the mass fraction of isotope i, and $P_i$ is its specific power, Watts/g). Accurate measurements of $R_i$ are required for an accurate determination of $P_{eff}$. Gamma-ray analysis programs such as FRAM and MGA are used to deconvolute the complex gamma-ray spectra such as shown by the solid line in Figure 3, but with these codes In many cases the uncertainties in resultant calculated isotopic composition are the largest contributor to the assay uncertainty. The ability of the TES microcalorimeter to resolve the individual Pu isotopic peaks will lead to more accurate isotopic measurements. A more detailed analysis of the improvement in precision for peak pairs with improved TES resolution can be found in reference [7].

Similar improvements in isotopic measurements can be obtained using TES alpha spectrometry. Measurements of the major Pu-238 alpha peak are used to determine the Pu-238 mass fraction for Pu with low Pu-238 weight percent, 0.01% - 0.7%, where the Pu-238 mass fraction is too low for an accurate measurement or where there is the possibility of U-238 interference [8]. Am-241, which has an nearly identical alpha decay energy, is a major interference. After a Pu alpha counting sample is prepared Am-241 starts growing in immediately from Pu-241 in the sample. The superior TES alpha detector resolution would be able to resolve the Pu-238 and Am-241 peaks and remove this possible interference. In fact the currently required radiochemical step separating Pu from Am would not be necessary if one were using the TES system. Since Pu-238, even at low mass fractions, is a significant contributor to item thermal power, a more accurate mass fraction measurement will lead to a more accurate calculation of thermal power, This is important in the case where the power of an NDA standard such as a CALEX standard that is to be used as a power standard for calorimetry is calculated from its isotopic composition and mass.

The high resolution of the TES alpha system, as shown in Figures 1 and 2, can be used to resolve the major alpha decay peaks of Pu-239 and Pu-240. Thus one can quickly determine the Pu-239/Pu-240 isotopic ratio from a low activity sample.

Another advantage of the gamma-ray TES system is that the higher resolution improves the peak-to-background ratio. This allows more accurate peak area determinations of weak peaks in a high background environment. such as the weak Pu K$\alpha$2 peak shown in Figure 4, where the simulated photon spectrum from a high-burnup PWR fuel pin as detected by a TES detector with a 50 eV FWHM resolution is shown overlaid with a simulated hpGE spectrum. The 100 eV natural line width of the fluorescent U and Pu X-rays generated by fission product beta- and gamma-rays such as Sr-90/Y-90 and Cs-137 compared to the much narrower Eu-155 105.3 keV line should be noted. From the Pu/U X-ray ratio one can obtain an estimate of the Pu content of a spent fuel assembly [9,10].
Currently the operation of TES cryogenic microcalorimeter spectrometers differs
from that of current technology. One is accustomed to directly observe the alpha- or gamma-ray spectrum on a computer screen in real time from a multichannel analyzer. The signals used for the spectrum shown above were processed off line using a signal-processing algorithm for each pulse.

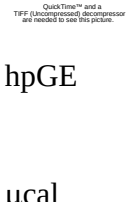


Figure 4. Simulated photon spectrum from a high-burnup PWR fuel assembly as detected by a 256-pixel TES microcalorimeter with 50-eV resolution and an hpGE detector. GEANT4/Translat™ simulation.

To be useful for practical applications and be equivalent to current data collection timeliness, real-time signal processing algorithms are being developed to process the data and provide a real-time spectrum and be capable of handling high count rates (100 Hz/pixel) while maintaining high resolution.

Another advance in the making this new technology practical is the development of dry cryogenic systems, not requiring liquid nitrogen or helium. These systems have been developed for integration with a TES spectrometer and are capable of maintaining constant sub 0.1 K temperatures for a week. This capability will allow the TES detectors to be used in analytical safeguard laboratories or in field operations.

## Conclusion

This paper has presented alpha- and gamma-ray spectrometry measurement results using cryogenic TES microcalorimeters. The great improvement in resolution obtained by using this technique can lead to improved isotopic measurements by separating currently unresolved alpha and photon spectral peaks. The Increasing sophistication of signal processing capability will allow the construction of large arrays of TES pixels to obtain reasonable count rates for gamma-ray applications with the goal of allowing for practical operation of such systems.

## Acknowledgements

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# Large volume and high sensitivity calorimetry for nuclear materials investigations

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***Abstract:***

*Calorimetry has been used for many years in the nuclear field (civil and military) but the knowledge of the applications remains very confidential. However the thermal characterization of the nuclear materials is a critical problem that has to be solved.*

*One problem is the management of the nuclear wastes coming from different sources. It is needed to accurately know their thermal activity in order to find the adequate storage mode and define the safety parameters of this storage.*

*Another problem that is easily solved with calorimetry is the quantification of radioactive elements, especially tritium in the military nuclear activities. As it is very important to trace these elements, calorimetry by measuring the activity of the elements is used to give very precisely the amount of the element in a container. In order to fit with the requirements of the nuclear research sector, SETARAM Instrumentation develops large volume calorimeters more particularly for the analysis of radioactive wastes and for the quantification of radioactive elements (e.g. curium, plutonium, tritium). In fact, the calorimetric detector has to be adapted to the size of the vessels containing the radioactive substances. These instruments are also used to measure the energy of hard gamma and neutron emitting sources in shielded containers. The large-volume calorimeters are able to accommodate 60- and 90-liter containers.*

*Thanks to their excellent sensitivity, the large volume calorimeters can accurately measure very low thermal effects (less than 10µW/litre, that is to say 10 nW/ml), which makes them powerful working tools for nuclear applications. The calorimeters, which conform to EC standards, can be placed in glove boxes under controlled atmosphere. The construction materials are chosen for their good fire resistance and the fact that they can be easily decontaminated. All the electronic unit ensuring temperature control, heating power and data acquisition, can be remotely positioned outside the "contaminating zone".*

**Keywords:** Calorimetry; NDA; Tritium quantification; waste characterization

## 1. Introduction

The use of the radioactivity generates no reusable radioactive wastes that must be stored with traceability and precise safety requirements. These wastes come from the industrial processes of the nuclear fuel cycle (extraction, transformation, exploitation and reprocessing), from research laboratories, from civil or military experiments and from biomedical uses.

In order to respect the regulations in force, it is then imperative to classify these wastes according to the intensity of the radioactivity as well as nature and toxicity of the product contained. Thus, wastes need to be screened and characterized by non-destructive means of measurement, the destructive means being intended for particular or exceptional controls carried out in specialized laboratories.

One way of obtaining the essential data about the intensity of the radioactivity, is the experimental determination by calorimetry. Very precise and highly reliable non-destructive measurement, calorimetry is able to classify the radioactive wastes and thus, to define their method of storage.

## 2. Features of the calorimeters

These calorimeters are built in order to measure with accuracy very weak thermal effects generated by high volume sample. The main features of the calorimeters are described in the following table:

| **DIMENSIONS** | | | |
|---|---|---|---|
| **Measuring and reference cells** | | | |
| | 270 | 390 | 680 |
| Internal volume | 15 liters | 60 liters | 90 liters |
| **General dimensions** | | | |
| Height/Width/Depth (mm) | 770/1040/850 | 960/1500/1000 | 1260/1500/1000 |
| Mass (kg) | 500 | 1000 | 1200 |
| **PERFORMANCES** | | | |
| Lower limit of detection (mW) | 0.25 | 0.50 | 1.00 |
| Measuring range (mW) | 0.25 to 3 | 0.5 to 13000 | 1 to 26000 |
| Sensitivity (µV/mW) | 70 | 160 | 155 |
| Signal stabilisation time (h) | 10 | 30 | 30 |
| Working temperature | Isothermal | | |

**Table 1:** main features of the calorimeters.

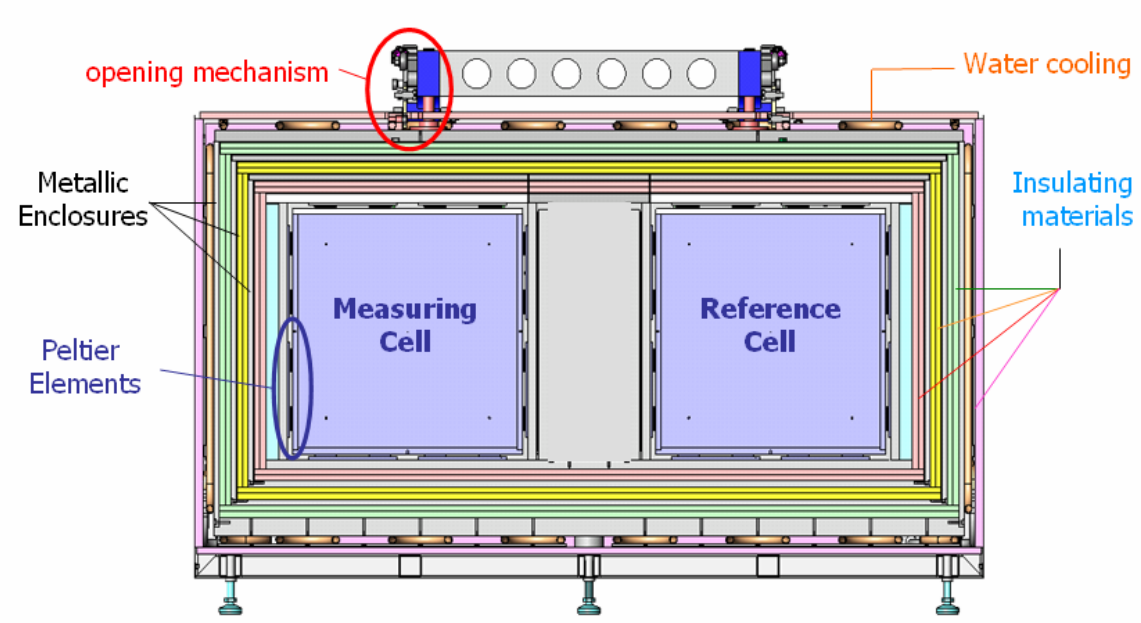



**Figure 1:** calorimeter cross section.

These calorimeters are differential with a measuring cell for a container loaded with the active product to be analyzed and a reference one for a container that is either empty or loaded with an inactive product. Heat flow measurements are made by Peltier elements connected in series that we use as heat flux meter sensors due to the Seebeck effect.

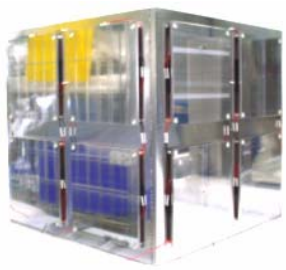


**Figure 2:** calorimeter cell.

These Peltier elements which were specially designed for this kind of application cover all the six sides of each cell so that virtually all the heat flux produced inside the measuring cell can be integrated. The heat flow corresponding to the energy evolved from the active product is measured as the difference between the measuring and reference cells heat flow.

The inside of the calorimeters is specially designed to avoid any thermal disturbance coming from the external environment. Thus, several metallic enclosures are disposed and between each enclosure, insulating material panels are put. Owing to this shape, we can ensure a perfect insulating of the calorimetric block and a very good thermal homogeneity.

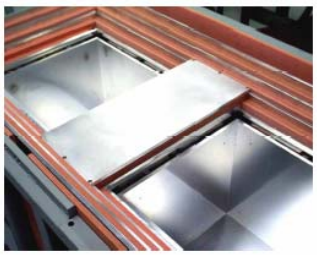


**Figure 3:** inside of the calorimeter.

Concerning the temperature regulating system, the calorimeters have two independent PID regulating loops managed by a Eurotherm controller. The first loop ensures the lid regulation with four platinum probes and the second one ensures the calorimeter block regulation with four others platinum probes. These two independent loops are necessary to improve the stabilization time of the calorimeters because when it is opened, the lid cools more rapidly than the calorimeter block so the lid-controlling loop will be able to compensate this phenomenon.

**Figure 4:** Eurotherm controller.

In order to heat the calorimeters, we use laminar heaters which are glued on four faces of the internal metallic enclosure.
Owing to this heating principle, we can ensure homogeneity of heating, a good heating speed and a high quality of the thermal contact.

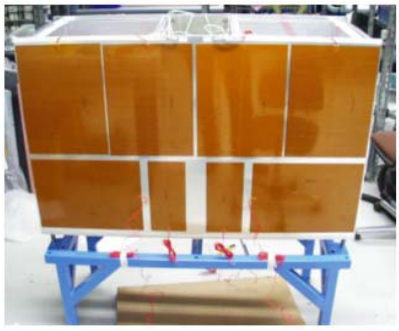


**Figure 5:** laminar heaters.

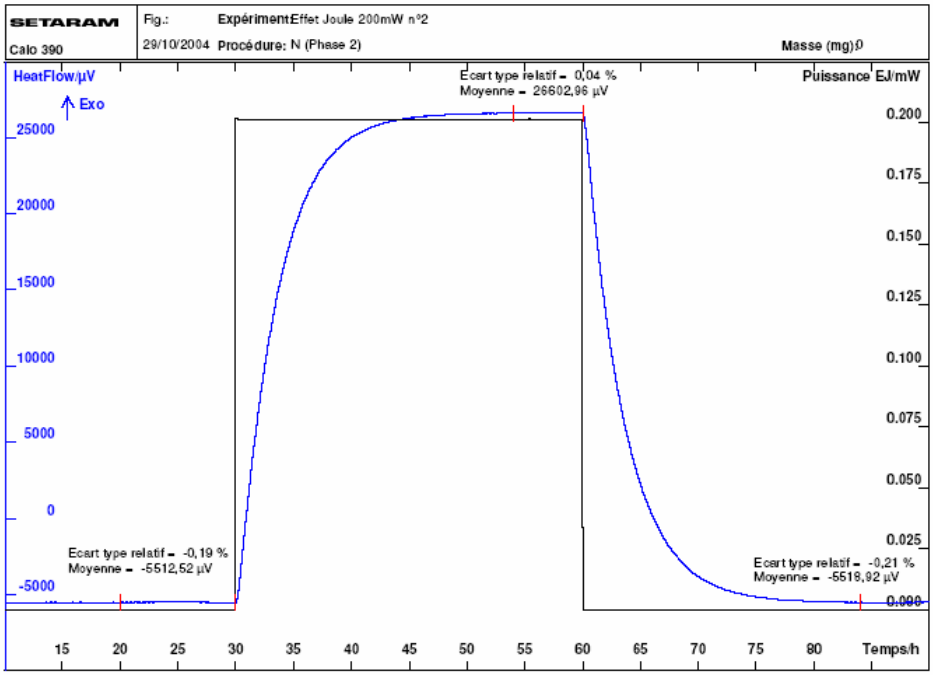
Finally, to improve the evacuation of the calories released by the sample, a cooling system made of copper tubes unit is fixed on the six faces of the external enclosure.
This system works in closed loop using a thermostatic bath, which controls the water temperature to the hundredth of degrees centigrade. Moreover, this water circulation creates a thermal barrier for the external disturbances.

**Figure 6:** water circulation system.

# 3. Example of application

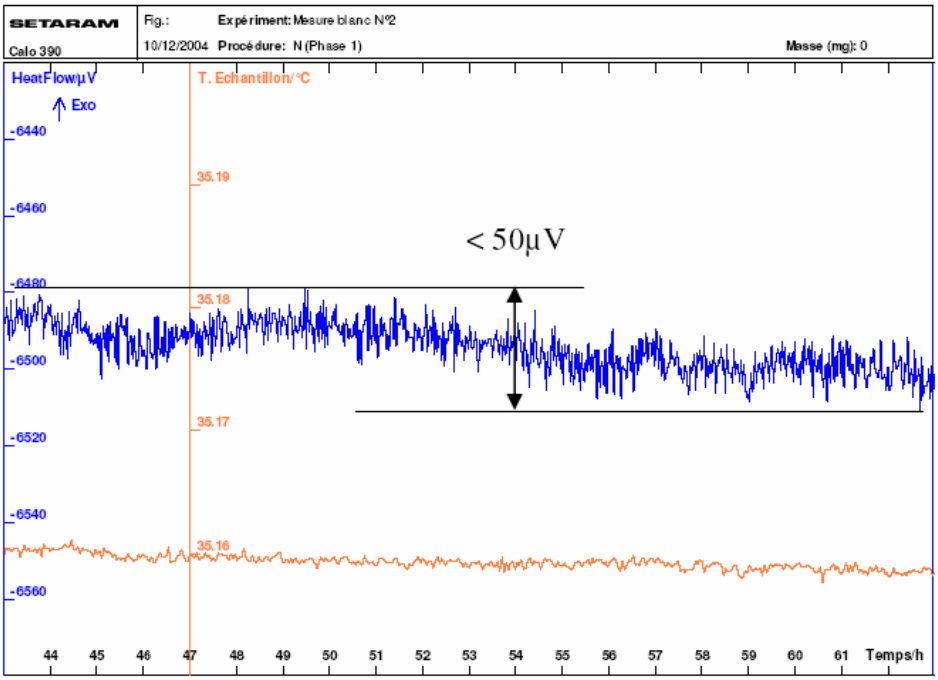
## 3.1. Tritium mass measuring

Before measuring a real sample, it is necessary to know exactly what the sensitivity of the calorimeter is. In order to determine this value, we use a Joule effect cell which generates a very accurate thermal power inside the calorimeter.




**Graph 1:** 200mW Joule effect.

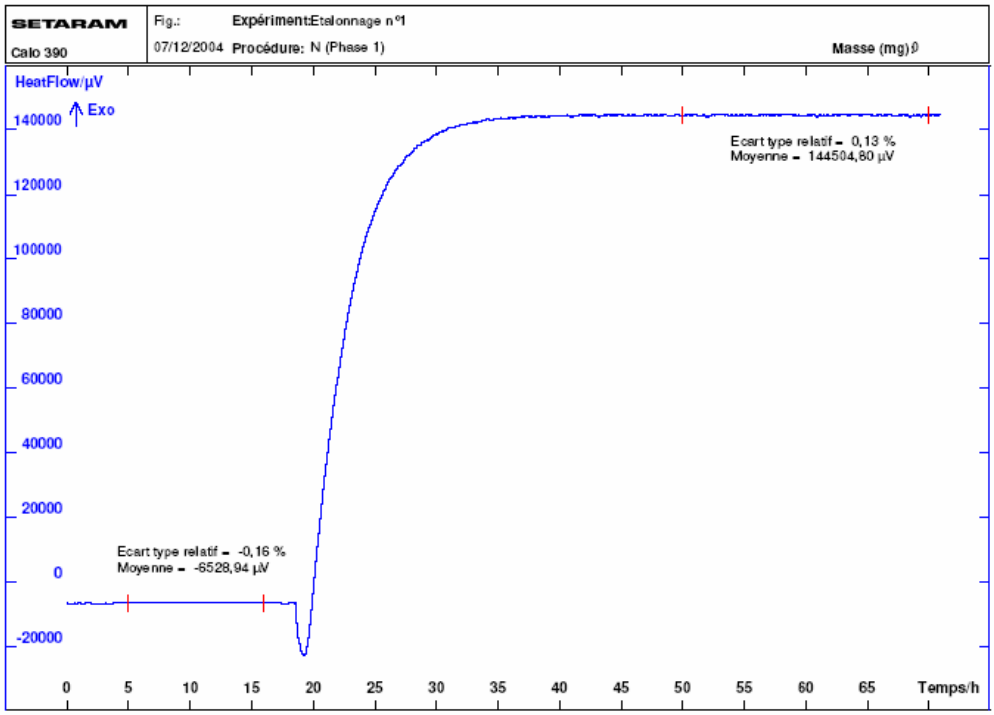
While a Joule effect of 200mW, the heat flow deviation is equal to 32000µV. Hence, the sensitivity of the calorimeter is 160µV/mW. The time constant of the calorimeter is nearly 3.5 hours.




**Graph 2:** baseline.

Moreover, the calorimeter is very stable with a baseline which long term noise is lower than 50µV that is to say lower than 300µW. The temperature inside the calorimeter is taken on the measuring cell and we can notice on the graph that it is very smooth with a long term noise about 1mK.

After having characterized the calorimeter, it is thus possible to make a measure on a real sample. The goal consists in measuring the real Tritium mass contained in a shielded metallic container.




**Graph 3:** Tritium mass measuring.

Once the calorimeter stable, the sample is introduced inside the measuring cell. The opening of the calorimeter causes a perturbation on the heat flow signal. After several minutes, the heat flow signal grows up and reaches the equilibrium after 20 hours. The heat flow deviation caused by the sample is equal to 150000µV that is to say 937.5mW due to the calorimeter sensibility. As we know perfectly the exact thermal power generated by one gram of Tritium, we can convert the heat flow in Tritium mass. Then, the result is 2.88 grams.

## 4. Conclusion

The large volume calorimeters are especially dedicated to the nuclear field owing to their sensitivity, their high capacity, their measuring accuracy and their reproducibility. Thus, the calorimeters represent powerful working tools in order to classify nuclear wastes according to their radioactivity which is directly linked to the thermal power generated by the sample. In the particular case of the Tritium study, the detection limit of the calorimeters is 1.5mg.

However, these apparatuses can be very useful in a lot of other applications where high volume samples are studied and where small energies are generated like in environment, biosciences or the batteries studies.

# Session 19

# Systems of Accountancy and Control (SAC)

# Current status of the implementation of SNRI regime in ABACC Regional System

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***Abstract:***

*During 2006, ABACC and IAEA have been discussing the conditions to initiate the application of a Short Notice Random Inspections (SNRI) regime, in January 2007, in Argentinean and Brazilian fuel fabrication and conversion facilities. The main objectives of this new safeguards regime is to achieve 100% coverage of the nuclear material transfers term of the mass balance equation during the material balance period, to confirm that the facilities are operating as declared and to make internal and external borrowing scenarios and abrupt diversion of nuclear material more susceptible to be detected through the introduction of randomized interim inspections and simultaneous physical inventory verifications. In addition, during these random interim inspections traditional activities required in the safeguards criteria, such as the verification of strategic points, the operator's measurement system assessment and records and reports auditing will take place.*

*The introduction of this new safeguards regime involves the implementation of some innovative elements such as the retention periods on nuclear material subject to be transferred, the operational mailbox declarations and fixed sampling plans. In this regard, the frequency and inspection effort and the required information on the advance notifications are the dominant concerns. All these changes imply arrangements and new procedures that have required consultations with the state parties/operators in order to establish the most adequate parameters. In addition, some particularities of the ABACC's Regional System were taken into account in order to avoid undue burden on the operators and to preserve the improvement in the effectiveness that implies the introduction of unpredictability in the planning and conducting interim inspections.*

*Taking into account that the characteristics and the operational schedules of the fuel fabrication and conversion facilities in Brazil and Argentina are quite different, the impact of the SNRI regime on each facility is analyzed in this paper and the coordination arrangements and parameters adopted on each case to meet the safeguards goals are presented. Finally, the current status of implementation is reported.*

Keywords: nuclear safeguards implementation; regional systems; experience in new domestic transfer verification methodology.

## 1.- Introduction

Since 1994, ABACC Regional System applies full safeguards procedures on all the fuel fabrication facilities in Argentina and Brazil. Within this type of facilities the two commercial facilities handle more than 2 significant quantities of nuclear material in a year, then the SNRI criteria can be applicable. One of them is in Brazil; this plant produces all the nuclear fuel assemblies of low enriched uranium required to refuel the light water reactors Angra I and Angra II. The other one is in Argentina; this facility produces the nuclear fuel assemblies of natural and low enriched uranium required by the on load reactors of Embalse and Atucha I.

Until 2006, the Safeguards Criteria required the verification of the inventory of nuclear material once a year and at least 20% of the nuclear material, per category type, involved in domestic and international transfers. Other objectives like the verification of strategic points, operator's measurement system, blending activities and other inventory changes, were normally addressed in connection with the PIV or interim inspections activities. In addition, provisions to cover the borrowing scenario have to be taken when applicable. Verification activities serving timely detection purposes are not foreseen in this type of facilities since none of these facilities operates with plutonium or high enriched uranium.

The operational characteristics of both fuel fabrication plants are quite different. The Brazilian facility is a conversion and fuel fabrication plant where all the production steps are integrated. The facility comprises two main process, one dedicated to produce UO2 powder and pellets from low enriched UF6 (up to 5% enrichment) and the other dedicated to fuel assembly production. The facility works by campaigns, normally two per year, with a maximum design capacity of approximately 170 tons of LEU. The major components of the flow term in the annual mass balance equation are the domestic shipments of fresh fuel to the nuclear power stations and the domestic receipts of UF6 and rejected fuel elements. The major international transfers are the imports of low enriched uranium as UF6 and U-Gd fuel rods and the exports of UO2 powder and pellets.

The Argentine facility is a fuel fabrication plant that receives the UO2 powder and produces pellets, fuel rods, LEU fuel assemblies and NU fuel bundles. At present, the design capacity is approximately 140 tons of natural uranium and 100 tons of LEU to produce 5000 CANDU type fuel bundles and 500 Atucha I type fuel assemblies in a year. To produce LEU of 0,85% enrichment required for fuelling the PHWR Atucha I, blending of LEU up to 5% is carried out with natural uranium at the facility. The major domestic receptions are natural UO2 powder produced in a conversion facility and rejected LEU fuel assemblies. The major domestic shipments are the transfers of fuel assemblies and fuel bundles to the on load reactors and, occasionally, scrap material is shipped to the conversion facility to be recovered. The major international transfers are occasional imports of LEU powder. This plant works on a continuous basis. Up to two receipts per month of UO2 powder and normally one shipment per month of Atucha I type fuel assemblies and CANDU type fuel bundles, are foreseen in this case.

In March 2006, during the IAEA-ABACC Coordination Meeting, the Agency informed that an approved SNRI regime had to be in place by the end of 2006 in order to be possible to start the evaluation of the fuel fabrication plants and conversion facilities applying these new criteria in 2007. Following this notification, ABACC coordinated consultations with the state parties in order to immediately initiate with discussions, considering that the new regime introduces additional burden on the operators and national authorities and requires special arrangements in order to adapt some particularities of the ABACC's Regional System, regarding channels of communications, coordination of joint inspections and triggering, verification of imported nuclear materials, etc.

Since May 2006, the main elements of the new regime were discussed among the parties. Comments, points of view and proposals where exchanged through e-mails and in September 2006, during a quadripartite meeting, the basis to implement the new regime were established.

Following this meeting, progress in the discussion of the formal documents was met and tentative dates for field trials were proposed.

## 2.- Pro and cons of the new regime

The inspection activities and methodologies considered in this new regime are aimed at achieving the following objectives:

a) To improve the safeguards effectiveness through the introduction of unpredictable interim inspections.
b) To make possible 100% coverage of the flow term of the annual mass balance.
c) To make the borrowing scenarios more susceptible to be detected.
d) To make false reporting more difficult to be concealed.
e) To improve the detection probability of abrupt diversion.
f) To provide a better confirmation that the facility operates as declared.

Even though this new regime is more effective than the present verification strategy, from the facility operational point of view, its implementation reduces the flexibility in the operational program, introduces undesirable death time in the availability of the feed material and final products and requires extensive and more frequent provision of data. Consequently, extra burden on the operators and national authorities is expected.

## 3.- Issues related to SNRI regime in the regional system.

The implementation of this new approach requires the introduction of changes in the current practices. In order to avoid duplication of efforts between ABACC and the IAEA, new arrangements regarding notifications, coordination of inspection activities and verification of international transfers have been discussed and agreed with the state parties.

a) Provisions to avoid the duplication of efforts.

The Quadripartite Agreement requires that duplication of efforts should be avoided whenever possible. In this regard, a joint inspection program was adopted as the most efficient strategy.

b) SNRI notifications arrangements

Under the traditional approach, the IAEA notifies in advance the regional system the inspection activities and the regional system is the responsible for the coordination with the state parties.
In the new regime the triggering organization assumes the responsibility to notify the other organization and the state party its intention to carry out a SNRI. The notification of the triggering organization has to be sent to the state party 24h in advance. Provisions to ensure the participation of the non-triggering organization have been adopted in a specific coordination guideline. It was agreed that the triggering organization will coordinate the inspection.

The identification of the inspectors, their documentation, the selected facility and the inspection starting time must be included in the notification to the State Party.

This arrangement is valid only for SNRI notification. All the other inspections will follow the traditional practice, according to the Quadripartite Subsidiary Arrangements.

c) Verification of international transfers

The General Procedures of the Common Accounting and Control System of Nuclear Materials (SCCC) established under the Bilateral Agreement for Peaceful Uses of the Nuclear Energy - signed between Argentina and Brazil- requires the verification by ABACC of all the international transfers of more than one effective kilogram. As the IAEA has the possibility to verify such transfers of nuclear material in the supplier or receiver country, this requirement could imply an undue burden on the operators and interfere in the coordination of the inspection activities.

The Secretary of ABACC has adapted this procedure in order to improve the coordination of the inspection activities at fuel fabrication facilities. As a result of bilateral consultations, it was decided that in commercial fuel fabrication facilities the international transfers will be verified at the moment the SNRI takes place. Considering that the imported material is normally used as feed material in the fabrication process, the remainder material not yet processed, available at the facility at the moment the SNRI takes place, will be verified in order to confirm the international transfer. In addition, retention period will not be requested for imported materials at the fuel fabrication plants and conversion facilities under SNRI regime.

d) List of item subject to verification.

The SNRI inspections are aimed at verification of nuclear material involved in domestic and international transfers and to cover the internal borrowing scenario. The nuclear material subject to verification will be included in a List of Inventory Items (LII). This List will be provided to the inspectors upon arrival at the facility.

As the SNRI regime deals with the flow term of mass balance equation, nuclear material corresponding to intermediate compounds are not submitted to verification. Only the nuclear material received and not yet processed and the nuclear material ready to be shipped, available for verification at the facility, shall be included in the list.

Nuclear material involved in domestic and international transfers, previously verified and still available at the facility, shall also be included in the list and identified. This material will be verified differently following the borrowing scenario provisions.

e) Verification of nuclear material

All nuclear material involved in domestic transfers included in the list and not yet verified will be verified applying a fixed sampling plan.

All nuclear material involved in international transfers, if present at the facility, shall be verified with medium detection probability at the same level of the fixed sampling plan.

All nuclear material included in the list and identified as previously verified will be verified for gross defect with low detection probability.

Each facility has a fixed sampling plan approved based on the annual throughput. This plan defines for each stratum submitted to verification, the quantity of items to be verified and the level of verification (gross, partial and/or bias defect) independently of the total population of items available for verification at the moment the SNRI takes place.

Consequently, regarding sampling plans, different criteria are applied for domestic transfers, borrowing and international transfer.

f) Operator declarations to support SNRI

Requirements to provide the information indicating expected dates for international transfers of nuclear material and the annual operational programs are clearly established in the Subsidiary Arrangements of the Quadripartite Agreement.

In addition to these provisions, the new regime requires frequent SNRI declarations to support the verification activities. In the traditional practice, all the operational and accounting information from the states parties are sent to IAEA through ABACC. In order to support the new approach, ABACC and the states parties have accepted that the SNRI declaration will be provided simultaneously to ABACC and IAEA, via encrypted e-mail.

The frequency of the SNRI declaration was established on a case by case basis, taking into account some operational concerns.

As an example, we can mention that in the facility that works with a continuous regime, a weekly frequency of SNRI operational declarations was adopted for UO2 powder received and fuel assemblies ready to be shipped.

g) Records and report auditing.

The joint auditing software agreed upon between IAEA and ABACC is applied to carry out joint book auditing activities. ABACC is responsible to provide updated accounting information. It was agreed that in case the non-triggering organization can not participate in the SNRI, the SNRI declarations will be checked for correctness and consistency against supporting documentation. A complete book auditing will be performed during the next joint inspection.

h) Retention periods

The retention period is the period of time while the nuclear material involved in domestic transfers can not be processed or shipped to be available for verification at the facility. As a general rule, 5 working days were adopted in the Brazilian facility and 6 working days in the

Argentine facility, introducing special provisions regarding quality assurance of the fuel assemblies in the last case.

## 4.- Current status

At present, draft documents for both facilities have been discussed in detail with both States Parties. Last versions dated 29-03-2007, were received by ABACC on April 2007 and submitted to Argentina and Brazil for final revision and formal approval.

For the Brazilian facility, a clarification from the IAEA side on the practical implementation of the 24 h advance notification to the State Parties during a day off is still pending.

In the case of Argentina, the conversion facility is not integrated with the fuel fabrication plant. Consequently, the national authority has requested that the nuclear material involved in domestic transfers between both facilities should be verified with the same criteria. At present, the SNRI regime has been proposed only for the fuel fabrication plant, while alternatives to apply a SNRI for the conversion facilities are being explored.

After the formal approval of the documents, a field trial will start. ABACC expects that in July 2007 the field trial will be running at least in one of the facilities mentioned in this paper in order to adjust with the operators and both national authorities practical details of the implementation of the new regime.

Still there are some differences between the IAEA side and the State Parties, regarding the format and the data that should be included in the SNRI declarations. This declaration represents an additional burden that must be assumed by the operators. The national authorities would like a less rigid format for these declarations while the provision of the relevant data of each item for verification purpose is assured. In this regard, the provision of scanned copies of the packing lists is much easier for the facility operators than typing the same data in a rigid format like an ICR. In addition, the introduction of special codes, can be source of frequent mistakes until the operator gains experience in their application. ABACC supports the proposal of the national authorities.

## 5.- Conclusions

From the overview presented in this paper, it can be observed that the introduction of this new methodology of verification of nuclear material involved in domestic and international transfer required from the national authorities and ABACC the modification of the methodology currently applied for international transfers verification and the introduction of relevant changes in the current practice, like channels of communications, coordination of inspections and the provision of safeguard information. Understanding that the new approach is more effective than the traditional one, the national authorities and ABACC have given support to this IAEA proposal. The discussions have been very fruitful and dynamic. As a result, the documents are almost ready for approval. It will take more effort and time consume to gain experience and to adjust some practical details at the field implementation. ABACC, IAEA and the national authorities should have an open mind and flexibility in order to successfully solve the pending

issues and to speed up the practical implementation of the new regime in both countries, Brazil and Argentina.

# The roles of a "strong" independent State System for Nuclear Material Accountancy and Control, the Hungarian practice

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***Abstract:***

*This article gives an overview of the quantity and quality of nuclear materials and the safeguards related research activities used and performed in the territory of the Republic of Hungary. Then it reveals the existing regulatory system, and the tools of regulatory supervision in the field of safeguards. Finally it draws a broad conclusion from non-proliferation point of view and summarizes the advantages of a strong national nuclear material accountancy and control system.*

**Keywords:** safeguards concept, national system, regulatory framework for safeguards, Hungary

## 1. Introduction

Hungary is a member state of both the Non-proliferation Treaty and the Comprehensive Test Ban Treaty, it signed and ratified the Comprehensive Safeguards Agreement and the Additional Protocol; the integrated safeguards approach was introduced in 2004. Hungary is a participating state of the Nuclear Suppliers Group and the Zangger Committee. At the end of 2006 the Hungarian Parliament ratified the INFCIRC-193 as an act, its introduction started in the middle of 2007. Hungary is a non-side letter state to the EURATOM Safeguards. The supervision of the national safeguards system (nuclear material accountancy and control, nuclear export-import control) in Hungary is conducted by the Hungarian Atomic Energy Authority (HAEA). The national prescriptions for executing the requirements derived from the above mentioned international legislations are issued in a decree of the supervising minister of the HAEA. This decree provides a frame for the data providing, licensing, supervision and on-site inspection activities within the country. The obligations undertook by the Republic of Hungary regarding the control of nuclear materials are fulfilled by a comprehensive supervision of nuclear materials. This comprehensive supervision is provided by an effective safeguards system, by the full use of the different legal tools.

## 2. The Hungarian nuclear profile

The use of nuclear materials in Hungary

- Preliminary nuclear activities
  - Nuclear materials in non-nuclear applications
    - Paintings, electric bulbs
- First steps
  - First radioactive source – 1954
  - Research reactor – 1959
  - Laboratories handling radioactive isotopes for medical, industrial, agricultural and scientific purposes
  - Zero reactors – 1959-1972
  - Training reactor – 1971
  - Nuclear power plant – 1978
  - Uranium mining (ore) – nowadays (extraction from water)

## 2.1. Hungarian fuel cycle

- Uranium ore selection – nowadays (yellow cake)
- Ore exported
  - No conversion
  - No enrichment
- Fuel assemblies imported
- Energetic reactors, research reactor, training reactor
- Spent fuel exported
- Interim storage of spent fuel (Spent fuel ponds, SFISF)
  - No reprocessing
  - Research for final depository

## 2.2. Nuclear facilities in Hungary

Hungary has four nuclear installations: the Paks Nuclear Power Plant, the Paks Interim Spent Fuel Storage Facility, the Budapest Research Reactor and the Budapest Training Reactor.

The Paks NPP consists of four units, which were commissioned in 1982-84 and 1986-87. The original electric capacity of each unit was 440 MW, which was upgraded by improving the efficiency of the traditional energetic equipment, thus currently the nominal electric capacities of the units are 467 MW, 468 MW, 460 MW and 471 MW respectively. In 2005 the Paks NPP provided 39,5 % of the domestic electric energy production by producing 13833,8 GWh electric energy, which may be considered as a fair mark compared with the previous years .

The building of the facility for the interim storage of spent fuel of the Paks NPP was made necessary by the uncertainty of their transportation back to Russia. The facility is intended to store the spent fuel from the reactor for an interim period of 50 years. One of the advantages of the modular system is that it allows the facility to be extended to take into account any increase in demand. Each module stores 50 spent fuel assemblies and their rowed situation allows to use a mutual reception building and a fuelling machine. In the present 11 modules 4267 spent fuel assemblies are stored, from which 500 assemblies are shipped from the nuclear power plant in 2005.

The Budapest Research Reactor is operated within the Atomic Energy Research Institute, and it is the most significant major equipment for scientific research in the area of physics in Hungary. The research reactor operates regularly from 1959, and it was granted further operating licence in 1993, after its reconstruction. The Atomic Energy Research Institute is responsible both for the operation and safety of the reactor. The most important area of application for the research reactor is the production of radioactive isotopes for primarily medical (diagnostic) purposes. The examinations of material structure necessary for the lifetime examination of the reactor vessels are carried out here, together with neutron-radiography and activation analytic research. The research possibilities significantly increased in 2000, when the cold neutron source started its operation that was a significant event for the research carried out in solid state physics. The research reactor has enough fuel for its operation in the next 4 or 5 years.

The training reactor of the Nuclear Technology Institution of Budapest University of Technology and Economics was established in June 1971, according to plans made in Hungary and by the help of Hungarian contractors. Its main function is to contribute to the training of the students and PhD students of Budapest University and other Hungarian universities in the fields of nuclear technology (reactor physics, reactor technology, nuclear energetics, radiochemistry and measurement technology), and radio- and environmental protection. Based on the request from the International Atomic Energy Agency, fellows from developing countries are trained here and the Institution regularly hosts scholars from the Agency.
When planning the Training Reactor attention was paid to the fact that it occupies a large area and that students – without certified nuclear qualifications – carry out measurements in the equipment. The protective systems in the reactor prevent nuclear accidents and the emission of radioactive material into the environment in case of the most serious possible failures or human errors.

## 2.3. Uranium mine

- Study of uranium density in the country (1949-1955)
- Mecsek mountains, close to south border
- Installation (1955-1957)
- First ore export to SU (1958)
- First yellow cake export - after chemical treatment (1962)
- Closing of the mine (1997)
  - ~ 40 years – 23 000 t U
- Uranium extraction from ground water
  - ~ 2-3 t U / year

**2.4. Other users**

**2.5. Nuclear materials**

- Natural uranium
  - In almost all facilities (chemicals, etalons)
  - Mecsek ÖKO Co. (extraction from groundwater)
  - Underground (20 000 tons)
- Enriched uranium
  - Fresh and irradiated reactor fuels
  - Enriched research materials (small amount, calibration sources, research samples)
- U-233
  - Few mg-s in research labs
- Depleted uranium
  - Only import (by-product of enrichment)
  - Container walls, radiation protection, counter-weights, ballasts
- Plutonium-239
  - Spent fuel (7-8 tons)
  - Pu-BE sources – (measurement of real content in IKI) (5-6 kg)
  - In smoke detectors (early application)
  - Etalons, metal samples
- Thorium
  - Gas-lighting – gas-skirt
  - GE metal-halogen lamps cathode-surface cover

2.6. Nuclear research

- Application of nuclear and radioactive materials
  - Industrial
  - Medical
  - Research
- No institutional research in the fields of
  - Fuel preparation
  - Fuel design
  - Fuel reprocessing
- Major areas in nuclear research
  - Training demonstrations
  - Instrument calibration

- Reactor safety analyses and reserach
- Particle acceleartors

## 3. Regulatory scheme for nuclear activities

**nuclear safeguards, nuclear export-import control, physical protection of NM, CTBTO**
**central registry of RM (sources & waste), transport, packaging, illicit trafficking**

## 4. Regulatory supervision of use of nuclear materials

In order to develop and continuously maintain the effective state system, the regulatory body for nuclear and radioactive materials utilizes the following elements:
(1) in the frame of safeguards licensing the regulatory body controls whether the arrangements to be elaborated by the organization possessing nuclear materials is able to meet the administrative and technical requirements, to support the effective supervision and to make the meeting of the objectives of on-site inspections possible;
(2) prescribing data provision obligations and processing the submitted reports it provides continuous supervision with regard to nuclear material accountancy, nuclear export-import and other nuclear activities requiring safeguards license;
(3) during on-site inspections the regulatory body verifies the information provided, and checks the installation and operation of the containment and surveillance equipment, and supervises the effective implementation of the safeguards related measures.

- For possessing nuclear material, for starting nuclear activity (first license)
- For modifications (modification license)
- For export-import of nuclear and dual use materials and equipment (export-import license)
- For termination of safeguards (exemption license)

- Requirement system (graded approach) is harmonized with large and small users
- 30 days deadline for regulator (may be extended with 30 days once more)
- Regulatory resolution (licensing and enforcement)
- Existing operators shall justify the compliance with new requirements
-
- Site characteristics (organization, layout, access, service parameters), owner, operator, sg responsible, activities
- Local accountancy system, archiving, data security
- MBA, KMPs
- Measurement, calculation and evalution methods of NM
- Accountancy procedures
- Technical properties relevant to sg
- SG duty system
- Surveillance and control arrangements
- Physical protection of NM
- Access of inspectors, obligations of inspectors

application 3 or 7 months in advance

- Any modifications, changes

  - comparing to those licensed in first license.

- The state subsequent to modification will meet the sg requirements.

- The nuclear facilities may apply for (nuclear safety and safeguards) licenses at the same time, but compliance with both requirement systems shall be justified

- International site inspections (data of 2006)
  - 52 man-days (IAEA)
  - 15 man-days (EURATOM)
- National site inspections
  - Escorting of international inspectors
    - drawing independent conclusion (records)
  - in 2006 (+ 17 man-days)
    - fresh core verification
    - fresh fuel unloading
    - supervision of elimination of damaged fuel (incident of April 2003)
    - inspection of imported equipment and materials

## 5. Conclusions

- Based on
  - Available technology
  - Research and technical capacity, experience
  - Stock of nuclear material
- Conclusions
  - Uranium ore extraction, no enrichment, no fuel preparation, no reprocessing, no final deposition (interim storage)
  - Reactors under safeguards
  - Good nuclear knowledge and experience (but not regarding proliferation)
  - Large amount of nuclear materials, but no technology for extraction or processing
  - Small amount in sites outside facilities
- Safeguards regulatory body
  - Same importance as nuclear safety (both for regulator and operators)
  - Safeguards aspects are considered in details
  - Strict deadlines according to national laws
  - Enforcement capability
  - Independent conclusion to Hungarian government

# NUCLEAR MATERIAL INVENTORY IN CASE OF EMERGENCY

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***Abstract:***
*Emergency situations for nuclear materials in the relevant facilities are provided for under French regulations. The decree issued on 12 May 1981 on the protection and control of nuclear materials currently specifies that "the Ministry of Industry can order a physical inventory of nuclear materials and compare it with audited records under all circumstances". These inventories can be stipulated for nuclear facilities, for example in the event of suspected theft, loss or diversion of nuclear materials. In such situations, operators must be able to check if all nuclear materials held are actually inside the perimeter of the facility and in the expected locations.*
*To test the organisation in the sites involved at operator and competent authority level respectively, exercises have been regularly carried out. These exercises have been used to validate the methodology as well as the composition of the various crisis committees and the relations with the different units involved.*
*In 2006, two operators of the Tricastin site, Eurodif and Areva NC were simultaneously involved. The scenario selected was based on the suspicion of the theft of one UF6 sample bottle associated with blackmail. During this exercise, a media pressure was exerted by professional journalists on the various implied entities: the two operators, the competent authority and his technical support body (IRSN).*
*After an overview of French domestic regulations, this paper describes the unfolding of this exercise.*
*http://esarda2.jrc.it/bulletin/autors.html*

**Keywords:** inventory, exercise, crisis, nuclear materials

## 1. FRENCH DOMESTIC SAFEGUARDS AND THE RELEVANT REGULATIONS

Approximately 300 facilities in France hold nuclear materials. These facilities cover the entire nuclear fuel cycle, from uranium mining to waste storage and are, for the most part, located on nuclear sites such as research centres, industrial complexes and nuclear power plants.

The French regulatory system on the protection and control of nuclear materials is based on Article L .1333 of the Defence Code and related regulations (such as Decree n°81-512 dated on 12 May 1981 and Order dated 16 March 2004). In this context, each facility must obtain an authorisation to hold or use nuclear materials and, for this purpose, must prove that all necessary provisions are taken to protect nuclear materials against theft, loss or diversion. In particular, an authorised holder must monitor and audit these materials. The data from all installations are assembled in an up-dated national accounting record held by the Institut de Radioprotection et de Sûreté Nucléaire (IRSN).

The Ministry of Industry is the body responsible for implementing these regulations, represented by the High Civil Servant for Defence within this Ministry. He is assisted in turn by IRSN, its technical support body. This control is essentially based on:

- Assessment of the measures taken by the licensee to guarantee the protection and control of nuclear materials. These measures are described in several files requested from the licensee (mainly the authorisation file);
- Regulatory inspections carried out by sworn, State-authorised officials.

In addition, French regulations provide for a physical inventory of nuclear materials and its comparison with the audited records being ordered by the authority under any circumstances (in the event of theft or suspicion thereof, for example). This type of inventory must be carried out within a few hours.

## 2. EXERCISES RELATING TO NUCLEAR MATERIAL INVENTORIES IN AN EMERGENCY

Operators, as well as the authority and its technical support, need an efficient organisation and appropriate training if they have to react quickly. Since 1993, exercises have been carried out periodically to test the potential organisation of a nuclear material inventory in an emergency. The main objectives of these exercises are:

- to test coordination between the various entities concerned (operators, the authority and IRSN);
- to clarify the role and the missions of all entities involved.

In addition, these exercises are used to achieve other, specific objectives, such as:

- training the emergency teams;
- testing the procedures defined specifically for that purpose;
- checking the efficiency of the operational resources (communications, measurements, etc.);
- estimating the time required to perform the various steps (activation of crisis centre, checking of physical protection devices, inventory of items and measurements in order to detect a difference between the declared amount of nuclear material and the material actually present. Only gross or partial defects are considered during an exercise.).

## 3. PREPARATION OF THE EXERCISE

Firstly, the facility or facilities to be involved in the exercise are proposed by the IRSN to the competent authority based on the specific features of the facilities and the objectives of the exercise. Different facilities are chosen in order to test the maximum number of operators holding sensitive nuclear materials (plutonium and enriched uranium). The three main operators CEA, AREVA and EDF are involved alternatively. On two occasions, CEA and AREVA facilities were involved together in the same exercise.

Next, the IRSN sets up a scenario. It is assumed that a certain quantity of nuclear material, whose chemical forms have to be specified, has potentially been stolen from a facility. The elaboration of the scenario is founded on a general context based either on proliferation, or on a malicious intent, or on both. Lastly, the simulation of media pressure constitutes an interesting option. It can be entrusted to professional journalists or to participants of the organizing entity.

To perform the scenario, it is interesting to base the choice of the articles and the possible repercussions on the security studies for physical protection or general accountancy. Another complementary approach is to examine the attractiveness presented by the various articles handled and transported in the installations (size, isotopic abundance, radioactive intensity…).

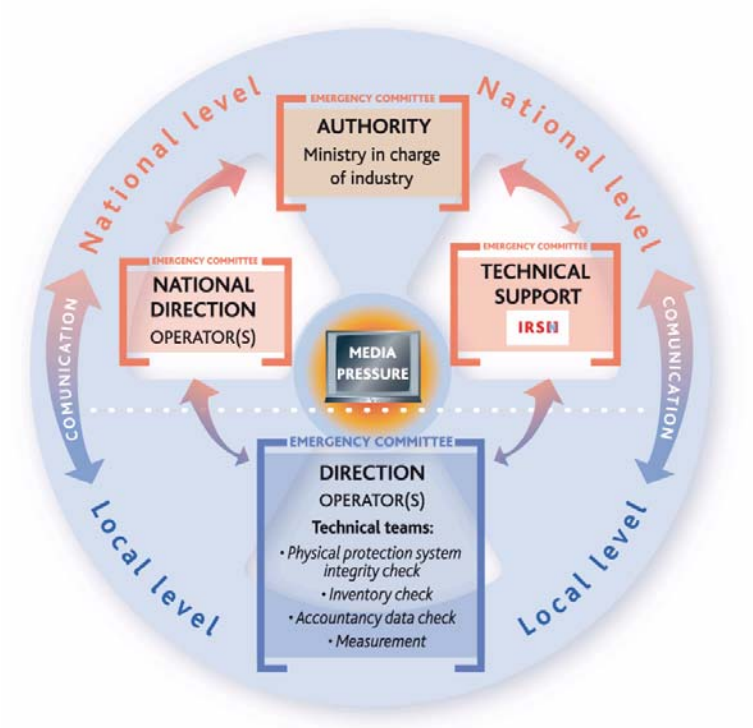
The practical items, such as the expected objectives of the exercise or its duration, are discussed with the entities involved. The operations of physical inventory must comply with the operating rules of the facility. The particular conditions related to radiation protection or to confidentiality are detailed there.

At the end of the preparatory phase, a convention setting the framework of the exercise is established.

The exercise is launched without prior notice over one day selected from within a fortnight chosen in advance with the operator.

## 4. EMERGENCY ORGANIZATION

During a nuclear materials crisis exercise, the organisation set up by the operator is very similar to the one designed for nuclear safety matters.




In case of a crisis, a national emergency organisation is set up, including the competent authority, its technical support (IRSN) and the operator at national and local levels.

The authority manages the physical inventory operations nationally and guides these operations based on the information available. It activates a crisis centre at its level to perform its mission.

IRSN sets up its own crisis centre in a dedicated room equipped with suitable communication resources. Its mission is to:

- act as the interface between the licensee and the competent authority;
- supply technical answers about the site and facilities to the competent authority based on the available documents;
- provide technical support to the competent authority, if necessary by calling on experts in the various divisions of the IRSN for a technical analysis;
- compare the local and national accounting records. The accounting records in the facility are compared with the monthly records of its inventory change reports held by the IRSN Central Accounting Office, which receives data from all the French facilities on a daily basis. This operation is intended to ensure that any possible theft, loss or diversion of nuclear materials in the facility has not been preceded or followed by any tampering with the local audit to cover up the illegal action and thus avoid early detection;
- inform the media on technical aspects of the crisis.

This crisis centre has its own communication resources and can communicate freely or with encryption (telephone and fax). It is connected to the crisis centres of the operators on the site(s) concerned with the exercise and with their headquarters.

It normally requires a crisis centre for the site management and nationally. In addition, other crisis centres are also activated for all facilities involved. The main missions of the management crisis centre are to:

- manage the various steps of the inventory;
- keep IRSN regularly informed on the inventory progress;
- reply to the authority or IRSN requests;

- inform the local media about the inventory development.

## 5. EXERCISE SEQUENCE

The exercise is launched by a fax sent by IRSN on behalf of the authority to the operator(s) concerned, who must acknowledge receipt. This fax specifies the type of nuclear material or the type of item (containers, for example) that may have disappeared. All crisis centres should be activated within a few hours following the alarm. The first measures taken by the operator may be to stop the processes and all nuclear material movements, close the site and request assistance from other specified services (radiation protection specialists, security guards, etc.).

Before taking the physical inventory itself, the facility should carry out the following preliminary steps:

- checking the physical protection devices (integrity of the fences, identification of alarms in connection with a potential theft, etc.).
- drawing up the list of all the items of nuclear materials held in the facility. It is possible to determine from this list the number of articles making up the inventory. This list of articles should also be compared with the audited records in the facility to make sure that the article database has not been altered;
- checking the accounting records in the facility and comparing them with the monthly records sent to IRSN.

Based on the characteristics of the inventoried articles, the checking proceeds from the most basic to the most detailed as follows:

- counting, identification and checking of seals on items concerned by the inventory taking;
- counting, identification and checking of tags on non-sealed containers of items concerned by the inventory taking;
- gross weighing of the preceding containers with a nuclear material content higher than a specified mass (value defined for the purpose of the exercise);
- gross quality checking (uranium or plutonium presence detection through physical measurements) of the preceding non-sealed containers (substitution by a dummy article should be considered);
- gross weighing and gross quality checking for sealed containers with a nuclear material content higher than the specified mass;
- fine quality checking of all containers with a nuclear material content higher than the specified mass (physical measurements of isotopic composition, Pu or U net weight, U enrichment).

The first checks should be carried out while the necessary resources for gross and detailed measurements are being established. Human and technical resources may belong to the facility itself or be mobile resources available to the site.

# PARTIE n°2 :

## 6. EXERCISE OF APRIL 5, 2006

## Sequence of events, experience feedback and, operator evidence

### 6.1. Launching of the exercise

The exercise started at 06:30 by the transmission of a fax to the central control room of Eurodif. The fax contained a photograph (shown below), with a message indicating that it had been transmitted by an individual, announcing his malicious intent.

Photograph of UF6 sample bottle transmitted by fax to Eurodif to launch the exercise

### 6.2. Chronology of events

- **From 06:30 to 07:30 :**
  - The shift manager calls the on-duty personal of Eurodif: Direction, safety, secretariat.
  - Call the on-duty personal of Areva-NC, as envisaged in the procedures.
- **From 07:30 to 08:30 :**
  - Involvement of the various national emergency committees.
  - Checks related to physical protection: closing access site, perimeter fences.
  - Editing of the bottle inventory listing, 995 bottles distributed in three zones of the plant are identified. Launching of the inventory in these three zones.
  - Detailed analysis of information on the photograph after transmission by email.
  - Realisation of the presence of similar bottles in the maintenance area on the installation close to Areva-NC. Involvement of Areva-NC's local emergency committee. Launching inventory at Areva-NC.
- **From 08:30 to 09:30 :**
  - Start of media pressure.
  - First official statement to the employees of installations and to the press.
  - First results of the inventory.
- **From 09:30 to 10:30 :**
  - First official statement to the press.
  - Estimation of the impact of a 1.5 kg of UF6 reject out of a volume of 10.000 $m^3$.
  - End of Areva-NC's inventory. No variation identified.
  - Media pressure on emergency committees.

- **From 10:30 to 11:30 :**
  - One bottle discrepancy identified at Eurodif.
  - Second official statement: both internal and to the press.
  - Media pressure of emergency committees
- **From 11:30 to 12:30 :**
  - Check of the bottle's data base. The missing bottle had been rejected, but its number had not been withdrawn from the data base that had been brought into service recently.
  - Media pressure on emergency committees.
- **From 12:30 to 13:00 :**
  - Conclusion: missing bottle without physical existence, no theft of nuclear materials.
  - *End of the exercise.*

### 6.3. Main experience feedback for the IRSN associated with this exercise:

- This exercise introduces two innovations: a context based on malicious intent and a scenario worked out in collaboration with a representative from each installation concerned. This way of proceeding appeared interesting and will be repeated in the future.
- The photograph that was used did not make it possible to establish if the bottle contained UF6 or not. It was thus necessary to carry out simultaneously the inventory on full and empty bottles. That showed the importance to preserve in a follow-up system all the data associated with empty packing.
- The exercise was launched with a photograph. But the poor quality of the transmission by telefax required the search for a higher quality datacomms network. As emergency committees are not equipped, this turned out to result in a loss of time. So, the identification in advance of a high quality and secured datacomms network close to each emergency committee appears advantageous. That type of connection is now available in the IRSN.

A feature of the exercise was the existence of a strong media pressure exerted by professional journalists on all the emergency committees. Then, during this exercise's evaluation, the journalists made the following remarks:

- The installation of an index that is understandable by everyone along with an indication of the rate of progress of the inventory is essential for having a discussion thread (% of articles or % of the inventoried masses for example).
- It can be very effective to set up preformatted sheet to answer efficiently to journalist's questions.
- It is very important that emergency entities involved exchange frequently some information to coordinate their answers to the journalist. In addition, each organisation should only communicate on theirs own field of competences.

# More than 30 years of safeguards at the Belgonucléaire MOX plant

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***Abstract:***

*The safeguards implementation at the Belgonucléaire MOX fuel fabrication plant in Dessel, Belgium, has for many years served as a model for similar nuclear installations. The experience gained at Belgonucléaire has helped both the European Commission and the IAEA to adapt their safeguards methods according to the requirements of an industrial process.*

*The plant was initially safeguarded by the European Commission but after 1977 it was jointly inspected together with the IAEA. In 2005 a decision was taken to close the facility after more than 30 years of operation. The facility has produced commercial MOX fuel for more than 20 years but at the end of 2006 the last production took place. The clean out phase of the facility has started while the start of the decommissioning depends on when a license can be granted.*

*This paper will review the safeguards activities carried out at the plant up until the present clean out phase. The experience gained during the lifetime of Belgonucléaire could provide useful feedback for other plants.*

**Keywords:** nuclear safeguards; review; operations

## 1. Introduction

Belgonucléaire was founded in 1957 with initial main objectives of performing research and engineering in the nuclear field. In the beginning the quantities involved were in the order of grams but after about 15 years the company started with MOX production in 1973. The foreseen yearly production was 3500 kg heavy metal as MOX fuel for Fast Breeder reactors or about 30 tonnes for LEU reactors. This remained basically the same for the entire lifetime of the plant, as at the end the production was around 35 tonnes of LWR MOX. The industrial production of LWR MOX fuel started in 1986 and in 2002 the plant had passed a cumulative production of 500 tonnes of MOX fuel!

The facility consisted at first of two sites, Belgo I / Mol which was located at SCK and consisted of Research laboratories and a pilot plant, Belgo II / Dessel, which was the Fuel Fabrication plant. The two sites became one on 22 May 1975.

The Commission has inspected the facility for about 50 years whereas the IAEA started to safeguard Belgonucléaire in 1977. This was a result after Belgium signed the Non-Proliferation Treaty in 1973 and the Subsidiary Arrangements to the Agreement which entered into force 21 February 1977.

Between 1978 and 1985 the throughput at Belgonucléaire was about 150 kg of Pu per year when they were producing fuel for fast reactors. Between 1986 and 1989 the throughput grew from 7.2 tonnes heavy metal as MOX (about 360 kg of Pu) to 35 in 1989 (about 1700 kg of Pu). This increase in throughput made it necessary to adjust the safeguards activities on a regular basis. One should also

remember that in the same period the public perception of the nuclear business changed as a consequence of the Three Mile Island and Chernobyl incidents. Also the modifications to the allowed dose uptake for nuclear workers affected the way operators could carry out their work.

The decision to close the facility was taken in 2005. The decision has been followed by several discussions with all parties, i.e. operator, the Belgian State, IAEA and the European Commission, on the way forward, taking IAEA and European Commission concerns into account.

## 2. Euratom Inspections

With the signing of the Euratom Treaty in 1957 the European Commission took on the responsibility of controlling the nuclear material at Belgonucléaire. In the beginning when Belgonucléaire consisted of a laboratory the control activities were carried out about once per year. The activities concentrated mainly on the accountancy, agreements between Euratom and external countries, and verifying the use of the material.

When the company started with MOX fuel production in 1973, the facility became increasingly important, and consequently the inspection effort increased. The material was received under seal and inspectors would be present during Belgonucléaire measurement activities, take samples, identify containers and, if necessary, seal such containers. About 16 samples were taken in one year for DA analysis and for attribute sampling all cans from the same receipt were checked for weight and that they were containing Pu.

Besides weight checks NDA techniques were introduced for verification in the form of a Geiger counter for Gross defect measurements and to verify the enrichment in $UO_2$ a mono-channel analyser together with a NaI/Tl detector was used. Later a multi-channel analyser was introduced together with a Ge detector and also a neutron counter to measure Pu. Neutron counting of fuel pins by comparison with a reference pin and random sampling and measurement of waste containers were also carried out.

In 1977 the estimated inspection effort with the above regime was 209 to 243 mandays plus 52 to 61 days at sites II and I respectively.

## 3. Joint Team Inspections

After Belgium had signed the Non-Proliferation Treaty in 1973 and the entering into force of the Subsidiary Arrangements to the Agreement on 21 February 1977, the IAEA started to get involved in safeguarding the facility. The Facility Attachment for Belgonucléaire was the first agreed one for Belgium and entered into force 1/12/78. Subsequently the PSP entered into force 21/3/80.

The first possible participation of the IAEA was in April 1977. When the IAEA started to safeguard Belgonucléaire all nuclear material was already under Euratom safeguards but for the IAEA there was a need to establish the beginning inventory of material. The IAEA performed a Design Verification in September 1977 and thereafter ad hoc inspections took place until the first JT inspection in March 1979. The inspection frequency was adjusted to be in line with the agreed timeliness of 2 weeks and for Pu a goal quantity of 8 kg was used. Key Measurement Points had to be modified to be in line with what had been agreed in the FA.

As a consequence of the timeliness component Belgonucléaire had to make the process area available for inventory verification activities to take place every two weeks which led to an increase in inspection activities for the operator. Besides these fortnightly inspections there were two Physical Inventory Verification (PIV) inspections per year. It was also necessary for inspectors to be present when significant quantities of material was received or shipped.

In the beginning the IAEA came as observers but there was a trend towards moving to One Man One Job (OMOJ). For the operator the worry was that this would lead to an increase of inspection activities when the lines are not stopped as two men can do more than one team in the same time and would therefore require two operators. The operator saw the positive side of this approach for activities when

the line was stopped for PIV activities and verifications at strategic points (later replaced with Short Inventory Verifications).

In the beginning of the JT inspections the throughput in the facility was about 150 kg of Pu per year when the facility was producing fuel for fast reactor. Between 1986 and 1989 the throughput grew from 7.2 tonnes heavy metal as MOX (about 360 kg of Pu) to 35 in 1989 (about 1700 kg of Pu). This increase of production made it necessary to continuously modify and adapt the inspection approach to the plants requirements/operating conditions and to introduce new instrumentation.

### 3.1. Equipment for JT safeguards

When the JT inspection regime started the IAEA accepted the Euratom equipment already present at the installation but they could perform independent tests of the equipment. IAEA equipment started to gradually enter the facility when there was a need to replace/update instrumentation.

The equipment used needed manual operation and included instruments for both neutron and gamma measurements. Neutron instruments included both passive (waste, small samples, fast breeder reactor fuel, fresh LWR fuel, $PuO_2$ powder and MOX pellets) and active (UF6) counting. The isotopic composition of Pu was evaluated with a CICERO which was replaced by the Pu meter together with the MGA code. There was a continuous need to look at the equipment situation in order to follow the evolution of the plant. Thus, to measure the incoming Pu an HLNCC was used and when new Cogema cans started to be used a UFBR-II was installed.

Other instruments were developed and introduced in order to improve the quality of the safeguards applied and to reduce the need to take DA samples. As a general rule one pellet DA sample was taken per timeliness inspection in order to control the quantitative verification of the operator's declaration of Plutonium quantity. A technique using a gamma detector to measure a pellet inside glove boxes in the process was introduced in 1987 and when it was combined with INVS measurements on bagged out pellets at a later stage, it reduced the number of DA samples of pellets to about 4 per year.

As the process area grew, it was also necessary to be able to move some of the equipment to different areas/rooms in the process and portable Ge detectors were introduced in 1991. At the same time a gamma detector for rod measurements was installed looking through a hole in the tray used to place the rods on during the neutron measurements. This made it possible to verify the isotopic composition simultaneously with the neutron measurement.

As a result of automating a blending station in 1989/1990 unsealed $PuO_2$ cans were present at the blending station at timeliness inspections which presented a problem and made it necessary to take at least one DA sample together with weighing of cans. New techniques needed to be developed for these verifications, reducing the need to take DA samples. The solution was high resolution gamma together with a neutron device installed in the process.

To maintain Continuity of Knowledge (CoK) an agreement was reached to use common seals and in 1982 there were discussions also to use surveillance. At first the equipment was not sophisticated enough, but new developments led to a surveillance system being installed in 1987 to keep the elements under dual C/S. The MIVS surveillance system was finally removed in 1992 as there were no longer any assemblies stored at the facility and the new LWR fuel assemblies were produced at the neighbouring facility FBFC Dessel (WBFP). When cameras were again installed they concentrated on the rod boxes (BAK) and the receipt of material. This reduced the amount of seals that needed to be replaced on the BAKs and the dose uptake for personnel.

When the facility started to concentrate on producing LWR fuel one rod could emerge from the rod scanner every 7 minutes. The rods were then loaded into a special tray and sent to an intermediate storage rack. Special transport containers were used to send the rods to WBFP. It was discussed that the inspectors would verify the material by 1) measuring completed intermediate storage trays and thereafter seal them and 2) observe loading of transport containers with subsequent sealing. This would have led to inspector presence about every three hours (time to complete a tray) for this operation. As a consequence there would be no extra handling of the material from the operator side

and it would also minimise the amount of rods left unsealed for timeliness inspections. This system was later replaced by an NCC in 1993 which could measure a BAK of rods instead of trays of rods.

In the beginning of 1986 the inspectors' office was supplied with a portable Olivetti PC! This facilitated the use of seals, helped in creating seals listings at the end of inspections and took over some of the tasks that had been carried out with HP41 and HP85 calculators. This also freed the HP85 to be solely dedicated to NDA instruments. After the introduction of computers the next step was to obtain the operator's data on a diskette and in 1988 the operator was able to provide the LII not only in a paper form but also in a computer readable form, a diskette. Further discussions resulted in the operator also providing flow and timeliness data in a computer readable form.

### 3.2. Samples for Destructive Analysis of Incoming Material

The incoming material was already under Euratom safeguards but the IAEA had no previous knowledge of the material. The IAEA wanted to sample this material but it was agreed that it was not necessary as this had been carried out at the shipping facility. Discussions on how to sample the incoming material continued and in 1987 a procedure to take random DA samples on incoming material when the $PuO_2$ cans were introduced in the process line was agreed. This change coincided with the introduction of the continuous inspection scheme and helped to improve the quality of the safeguards.

### 3.3. Timeliness Verifications and Flow Inspections

The timeliness component made it necessary for the inspectorates to be able to conclude that no abrupt diversion had taken place every two weeks. The nuclear material was therefore verified at strategic points every fortnight and the operator handed over a detailed listing of nuclear material representing the material quantities in the different operational units. The inspectors drew up a sampling plan of the material at the strategic points basing the stratification of nuclear material by weight and type to optimise the sample size. These controls were carried out Monday mornings and involved a 4-hour stoppage of the plant. Besides the fortnightly inspections there was also a need for the inspectorates to be present at every receipt or shipment of significant quantities of material.

The FA foresaw flow inspections as well and discussions started in 1981 to add one flow inspection week per month. For Belgonucléaire this resulted in NDA on receipts and finished material, flow verifications as appropriate, seal breakings and measurements on SNR bundles. At this time the assemblies were produced at the facility but later, when they moved over to LWR fuel, the production of assemblies was moved to WBFP. The flow inspections were implemented in the beginning of 1982 and led to a permanent presence of inspectors in the Mol area.

In 1983 the flow measurements developed further and resulted in more NDA measurements. The new scheme was tried before a planned shutdown of the facility in 1984 and was in line with the agreed FA. Improvements of the flow inspections and taking them into account led to discussions on the timeliness component and a move from two weeks to four weeks. The shutdown in 1984 was a result of a contract with Cogema – Fragema and CEA for Belgonucléaire to produce large scale MOX fuel for French thermal reactors. So far the process line had been stopped and cleaned out before every strategic point verification but when the facility started to produce MOX for LWR the operator wanted to avoid these production stops. The operator was worried at first that one inspection per month would take more time for them than two strategic point verifications carried out until then. The new scheme did result in an increased presence of inspectors, i.e. more time spent with the inspectors, but more importantly it meant a reduced stoppage time of the process as it would only be shutdown every 4 weeks instead of every 2 weeks. At this time a stop took 1½ - 2 hours. Compared to the old scheme the expected effort would only be slightly increased, from 80 – 100 days to expected 100 – 132 days per organisation.

The new approach consisted of a Short Inventory Verification (SIV) every 4 weeks, starting on the Sunday with preparation of sampling plans based on operator listings. Thereafter, there were two days of inspection activities on the Monday and Tuesday during which timeliness verifications were carried out using tag checks, HLNCC measurements in the stores, other NDA measurements on pellets, rods, assemblies and waste, and weighings and seal verifications/replacements. For non-SIV weeks a 1½

day flow verification inspection took place on Wednesdays and Thursdays. The first day was devoted to preparation and the second day for verification activities. The activities consisted of HLNCC measurements on received or unsealed cans with subsequent application of seals, NDA on Belgonucléaire quality control accepted pins and unsealed assemblies, with subsequent application of seals. The new scheme led to a greater safeguard penetration of the operators system with a situation where at any time a large part of the inventory would be verified and under seal. Also, the higher frequency of inspections led to better follow-up of material movements.

As the throughput grew so did the inspection effort and inspectors were eventually present from Tuesday to Friday with SIV activities in process carried out on Wednesday mornings. In order to start producing the LWR MOX fuel on a large scale the process was modified to 35 tonnes per year of MOX fuel in 1988 and a 2$^{nd}$ pellet manufacturing line installed. For the inspectorates this change involved a production of master blends that if not sent straight to the process, could return back to storage where they needed to be verified and sealed. Discussions to install inspector equipment in the process line started and the operator also started to provide the inspectorates with more detailed information on the material in the process line. As a result of the increasing production a continuous inspection regime started in 1988 as the JT started to be present also on Mondays. Previously the operator carried out internal transfers alone on Mondays, including detachment of seals. In December 1988 the operator asked the inspectorates to be present already as of Monday morning. After the introduction of the continuous inspection regime the material in the intermediate storage area was all under seal.

The planned capacity of 35 tonnes heavy metal per year was reached in June 1989. Inspectors were then needed to verify transfers of $PuO_2$ and MOX cans from store to process, to verify newly produced MOX cans, to verify receipt measurements and twice per day for loading of BAKs. As a consequence one inspector was needed for 5 days and one for 3 days every week from each organisation plus extra effort when SIV and PIV activities were carried out.

In April 1990 the SIV was moved to Saturday morning as problems had been encountered during the SIV activities and considering the time pressure on the activities as the operator was eager not to delay the start of production. The movement to Saturday mornings also led to a reduced quantity of green pellets, a reduced quantity of material in the oven and no material in the rod filling station.

### 3.4. Physical Inventory Verifications

Both the FA and PSP papers foresaw two PIVs per year. For the operator the PIVs took about 2.5 weeks including two weeks of preparation and the timings were such that they were carried out during the stops for summer break and Christmas break. As two PIVs per year took a considerable time to prepare and execute the operator asked to reduce the PIV frequency to once per year in 1980. This should be sufficient according to a study by SCK and CCR-ISPRA. However, the inspectorates were not ready to move to once per year at this stage. The operator reiterated this request on a regular basis and in December 1989 the operator expressed a definite wish to stop holding two inventories per year as the stops were too costly. A study was carried out by Euratom that showed that it would be possible to draw sufficient safeguard conclusions from one PIV per year combined with the monthly timeliness inspections which meant the end of two PIVs per year.

## 4. NPA Inspections

### 4.1. Change of inspection approach: 1994 to 1997: beginning of the NPA regime

A review of the Joint-Team inspection approach, being simultaneous continuous inspections by both the IAEA and Euratom, was initiated in the early 1990s. In 1992, a New Partnership Approach (NPA) was formulated between both inspectorates. One of the aims of the NPA was to achieve manpower savings for the IAEA in Euratom countries.

In the Joint-Team, all fuel measurements were performed manually in the presence of inspectors from the IAEA and Euratom. The transition from Joint-Team to NPA, where the IAEA would be present less frequently, involved the introduction of unattended, continuously operating measurement systems, with appropriate containment and surveillance devices (C/S) - independently verifiable optical-electronic sealing systems and video surveillance. Under these NPA arrangements, Euratom

performed inspections on a weekly basis – continuous presence Monday to Friday during normal daily working hours (Euratom inspections). The IAEA inspected for one week once a month for a short inventory verification (SIV), in order to evaluate the data collected from the unattended measurement stations, to verify the electronic seals (VACOSS) and to review the surveillance system [NPA inspections]. New procedures and working arrangements were developed to incorporate all these changes.

The NPA system for Belgonucléaire was conceived and equipment was installed from 1994 to 1996. Both equipment and procedures were extensively tested from 1996 to 1997, before coming into routine operation in October 1997.

## 4.2. Equipment for NPA safeguards

Continuously operating systems were designed and installed for the simultaneous measurements of gamma and neutron of $PuO_2$ powder cans on their receipt in containers (KOKER). The KOKERs were identified by surveillance cameras and measured before entering the store location. After measurement, the KOKER was moved to a storage pit and a shielding plug was placed on top. A VACOSS seal was applied to each storage location by a Euratom inspector.

A Neutron Coincidence Collar (NCC) system placed around the BAKs containing MOX fuel rods was used to measure their contents. The BAKs were identified by surveillance cameras and measured at the entrance to the store. VACOSS seals were attached at each end of the BAK in party line mode by Euratom inspector. The BAK was then transferred into the store, ready for shipment to a MOX fuel assembly facility.

Provision of duplicate data acquisition systems was the design philosophy common to all the measurement and surveillance systems to ensure continuity in case of equipment problems. The neutron and gamma measurement systems were in two cabinets accessible only in the presence of both inspectorates. Branching to a third cabinet enabled access by Euratom at all times without IAEA presence. For the surveillance systems, duplicated recording systems were installed within one cabinet. VACOSS seals were used, in combination with the surveillance systems, to complete a dual C/S system for both input and output stores.

- Receipt Measurement Station for $PuO_2$ Powder items (KOKER)s
  This unattended station consisted of both neutron and gamma measurement systems operating simultaneously. The neutron signal from a modified UFBC was split, and fed through duplicated JSR12 electronics. An adjustable support for the Ge detector allowed measurements, through holes in the UFBC, at five different positions on the KOKER corresponding to the five cans inside. The data from both neutron and gamma systems were stored on primary and secondary computers in separate sealable cabinets.

- Measurement Station for boxes of MOX fuel rods (BAK)s
  The NCC was installed on a movable trolley which allowed the detector to be placed by the operator, under surveillance, around the BAK for measurement. The electronics were similar to those for the neutron part of the receipt station.

- Surveillance systems
  Two different systems were used, being the MOSS (Multiple Optical Surveillance System), with a UNIPLEX back-up system. Eight cameras were installed for the KOKER store and BAK store. There were two cameras in the KOKER store; one used for recording the KOKER identification number, the other used to have an overview of all the storage positions and of the measurement detectors. There were six cameras in the BAK store. One was mounted on the BAK loading machine to read the BAK identification number. One provided an overview of the measurement station and the entrance to the BAK store. The remaining four cameras were needed to provide a complete overview of the open roof area of the BAK store, and of the storage locations.

- Sealing Systems
  Around 300 VACOSS seals were supplied. Each BAK was sealed with two VACOSS seals, one at each end – these were joined by a communication cable to facilitate verification from either end of the BAK, since only one end was accessible from a storage location. Every KOKER storage

position was sealed by one VACOSS seal between the shielding plug and the storage location body.

### 4.3. NPA procedural changes

Use of common seals by Euratom without IAEA presence was a significant step forward in working arrangements under NPA. These arrangements applied to an intermediate buffer store mainly used for cans of primary and secondary blend powder. During weeks when Euratom was present alone, any new flow of cans into the store would be verified and sealed with a 90% detection probability for Plutonium, using a one-method sampling plan, by an HLNCC.

In the week when the IAEA was present with Euratom, (NPA inspection), all the new flow into this store since the last NPA inspection was evaluated. For this material, already verified and sealed by Euratom, the IAEA agreed to apply a 2-method sampling plan with 50% detection probability, using an HLNCC and a CdZnTe or NaI detector for 30 second gross defect measurements. Thus, only a limited number of storage vaults were reopened for these re-verifications. Thereafter, all the Common seals placed by Euratom were accepted as valid. In addition, in order to evaluate the sealing system, Common metal seals which remained unbroken from one NPA inspection to the next were replaced with a 20 % detection probability. These were the compromises achieved to develop a workable procedure.

The operator also was required to make improved and more comprehensive declarations on movements of nuclear material. These principally concerned the movements of the boxes of MOX fuel rods into and out of their storage area, with precise dates and times. This enabled independent checks of his declarations by the inspectorates' neutron measurement and surveillance equipment.

### 4.4. Project Progress

Procurement of equipment began in 1994 and installation took place during the following two years. This included cabling for cameras, neutron and gamma detector systems, and associated equipment cabinets.

Field testing of equipment began on 28th May 1996, initially for a 3-month period. Towards the end of the period, an evaluation indicated that failures needed to be corrected and equipment reliability improved. Another evaluation after a further 3 months showed considerable improvement. It was concluded that the NPA arrangements could be implemented, although further discussions were needed on the recovery actions in the cases of equipment failures when the IAEA was not present. Thus, it took until October 1997 for routine operations to be approved by the Joint Euratom-IAEA Liaison Committee.

Key features in the successful implementation of this new NPA inspection scheme were regular working level contact meetings, well conceived systems including duplication of data acquisition when possible, and an adequate, well-planned field test period with detailed evaluation reporting.

### 4.5. Costs and Savings

From the perspective of inspection mandays, the estimated saving under the NPA inspection scheme at Belgonucléaire, from the IAEA side, was estimated to be a reduction to 99 mandays per year from their reference of 410 mandays in year 1990. The cost of the equipment installed and used to achieve this saving was of the order of one million €.

## 5. Further Development of the Inspection Scheme

After the implementation of the NPA inspection scheme Euratom had two inspectors present on a continuous basis. The main inspector concentrated completely on following the movements at Belgonucléaire whereas the 2$^{nd}$ inspector assisted but also inspected the neighbouring facility WBFP to which the final product of Belgonucléaire, MOX rods, were shipped. At WBFP they were assembled into the final MOX assemblies.

Further technical improvements made it possible also for Euratom to make savings in terms of inspector presence. The data acquisition computers were gradually replaced as they were getting old and the need for intervention occurred frequently due to breakdown. Also the data acquisition and evaluation software systems were upgraded and a FAST surveillance system was installed which meant that the time needed for surveillance review of Belgonucléaire was drastically reduced.

In 2002 a network was installed that transmitted the data from the receipt station and the BAK station directly to the inspectors' office via 5XP netscreen firewall boxes. The data was transmitted to a computer equipped with a 5XP box in a common sealed cabinet in the inspectors' office. The data were then copied to the PC's in the inspectors' office. This made it possible to check the working order of the equipment without entering the facility and to evaluate data, thus in case of problems the operator could be notified immediately. Also, the information from the FAST system was transferred to the inspectors' office with the same advantages. An inspector inspecting a neighbouring facility could make sure that the Unattended Measurement System (UMS) at Belgonucléaire functioned properly and, in case of need, an intervention could be staged after informing the Federal Agency for Nuclear Control (FANC) of the Belgian State. Both the Commission and the IAEA had inspectors in the area every time an inspection was planned in neighbouring facilities.

When studying how to make savings at Belgonucléaire it was important to include WBFP in the equation as there is also a UMS which is used to measure MOX assemblies. A risk of equipment failure at WBFP had to be minimised as a failure could lead to re-measurements being needed with increased dose uptake for the operator as a consequence. Before the computers were replaced at WBFP there had been several break downs but once the computers were replaced the situation was stable without any further reliability problems occurring.

As a consequence of the technical improvements the first significant reduction of Euratom inspector presence in the Mol area took place in the beginning of 2003, when it was decided that one inspector could cover Belgonucléaire (and WBFP) unless there was a shipment of MOX fuel planned from WBFP. Gradually the situation evolved so that the inspector present for Belgonucléaire would also cover the shipment at WBFP after an agreement with the operator at Belgonucléaire. A further reduction took place in the beginning of 2004 when it was decided to start the inspection at 13h00 on Mondays instead of in the morning, thus allowing the inspector to travel to the installation the same day. This would reduce the recuperation generated by the inspectors. In the meantime a study was undertaken to look at reducing the presence of Euratom to that of the IAEA, i.e. one week per month. The major consequences of a reduction of Euratom presence to the same level were the following:

1. The operator would have to detach common seals on locations in the intermediate storage area when material was needed for production. The knowledge of material stored in an opened vault would thus be lost for the inspectorates and therefore all material in an opened vault would have to be considered as having been transferred to process, even if the material remained in the storage vault. It would therefore be included as material returning from process during the next inspection and verified together with the normal material returning from process.
2. There would be no need to change the way $PuO_2$ material was received or the way measurements on the material were carried out. The modification was that the operator applied VACOSS seals on the storage positions into which the KOKERs were lowered in order to place the area under dual C/S.
3. For the BAKs the operator attached VACOSS seals prior to placing them in the storage channels so that they were under dual C/S. The operator would also detach VACOSS seals if there was a need to open a container.
4. VACOSS seals to be attached would come from a tray with seals ready to be used and detached VACOSS seals would be placed in a tray for later verification by the inspectors.
5. A further consequence was that the operator would have to compile information needed for the inspectors to update the database used by the inspectors to reflect the situation of sealed material inside the plant, with information of material movements inside the plant, so that it agreed with the real situation inside the plant. This information included a list of times and positions of detachments of Common seals, times and places of detachments/attachments of VACOSS seals, measurement times of received KOKERs, if any, measurement times of BAKs and shipments of BAKs to WBFP. This information would then be confirmed by verifying the VACOSS seals and performing a video review during the course of the inspection.

The study indicated that the reduced presence of inspectors would mean fewer measurements on cans going to the intermediate storage as Euratom verified all movements into/out of the store during the "flow" weeks using an HLNCC, or later a Multiplicity Counter. Euratom had in the mean time lowered the detection probability to the internationally accepted one, 50%, for the monthly MOX so there would be no extra measurements during the NPA week besides the normal measurements required by the IAEA. The negative impact was that all channels in the intermediate storage area would have to be checked and sealed during the same.

In June 2004 a meeting took place with the operator and the FANC and as of August 2004 also Euratom reduced its presence to one week per month. This transition went very well as the operator cooperated fully and provided all necessary information to the inspectors. The timing of verification activities was agreed and performed when other activities had to be carried out in the concerned areas so as to minimise the impact of the activities.

## 6. Closure of Belgonucléaire

A decision to close the facility was taken in 2005 and as a consequence the last fabrication campaign was completed on 15 August 2006. This had a significant influence on the inspection scheme and during the second half of 2006 the inspections were further reduced as the activities inside the plant decreased. The last Saturday SIV took place in November 2006 and as the vast majority of the material was placed under seal also the PIV in December 2006 could be done with a reduced number of inspectors from both organisations. The majority of the rest material present in the facility after the commercial fabrication campaigns were completed has been processed to rods and will eventually be shipped to La Hague for reprocessing. The material left in the facility is in the form of cans in the intermediate storage area and hold up in the process areas. The facility is awaiting the Royal Decree before the decommissioning activities can start and for the time being inspections will still be carried out on a monthly basis. The inspection activities consist of a C/S review as long as rods are present in BAKs and to ascertain that no production is carried out plus measurements on MOX cans in case material is recovered or repacked. For waste, the documents are controlled and gross defect measurements carried out. Once decommissioning starts the idea is that the waste will be shipped to the neighbouring facility Belgoprocess which has received the normal waste to date. The change in the status of the facility and the decreasing inventory has made it necessary to hold regular discussions between the operator, FANC, IAEA and Euratom on the way forward and the inspection activities are modified according to the status of the facility on a regular basis.

## 7. Conclusions

Belgonucléaire was in the forefront of not only MOX production but also in terms of how a small to medium sized MOX plant should be safeguarded. From the time when it was a research facility to the present status of the plant the safeguards approach had to be continually adapted for the benefit of not only the inspectorates but also the operator. The inspection scheme developed from yearly inspections via continuous JT inspections to NPA inspections and finally presence of inspectors one week per month. The short inventory verification schedule, which was introduced to cover the timeliness and consisted of controlling the complete inventory of the plant, started with inspections every two weeks. The change in the timeliness component and the need for inspection activities not to interfere too much with the commercial production led to SIV inspections being held on Saturday mornings once per month as of 1990. Neutron instruments were built into the process and improved the conditions for SIV activities. The introduction of computers simplified some aspects of the inspectors work considerably and made the installation of unattended measurement stations possible. These measurement stations were essential in making the NPA scheme possible. The reduction for Euratom was possible not only due to technical improvements but also thanks to the operator who co-operated fully and maintained good records of activities carried out during weeks when no inspectors were present. Without this it would not have been possible to quickly get a complete picture of the situation of sealed material inside the plant and get the inspector database updated. All changes were possible only in close collaboration with a committed operator and the excellent co-operation between not only Euratom and the IAEA but with all parties involved including the operator and the Belgian State.

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# Session 20

# Environmental Sampling & nuclear forensics

# Future Requirements for the Analysis of Environmental Samples and the Evaluation of the Results

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***Abstract:***

The Environmental Sampling programme of the IAEA plays an essential role in the process of determining whether States' declarations are correct and complete, in which includes assuring the absence of undeclared nuclear materials and activities. In order to properly assess the consistency of environmental sampling results with the declarations of a given State, it is of utmost importance that the Safeguards Analytical Laboratory (SAL) and other laboratories of the IAEA Network of Analytical Laboratories (NWAL) maintain a robust variety of analytical methods, perform measurements of the highest possible quality, maintain a high sample processing capacity, and provide results in a timely manner. Therefore, it is critical that the capabilities and capacities of SAL and the other network laboratories be improved and extended as much as feasible. The above issues will be discussed in light of the foreseen workload and new technologies.

**Keywords:** environmental sampling, safeguards, analysis, evaluation

## 1. Introduction

The IAEA carries out verification activities to support the drawing of safeguards conclusions regarding States' adherence to their safeguards obligations. This includes the provision of assurances that a State's declarations are correct and complete, i.e. that there was no diversion of declared nuclear material and that there are no undeclared nuclear materials or activities. In order to draw a soundly-based conclusion on a State's nuclear programme, the IAEA, inter alia, evaluates the declarations provided by the State and compares its consistency with information derived from in-field verification activities and obtained from open and other sources. There are currently six analysis/evaluations units in the Department. They cover areas like Open Source Data Collections and Analysis, Satellite Imagery Analysis, Trade and Technology analyses, State Declared Information Analysis, Material Balance Evaluation and Trend Analyses, and Environmental Sampling Evaluation.

The results of the various evaluations for a specific State, together with other information and knowledge obtained from different sources, are recorded periodically in a State evaluation report (SER). This is a key document as it forms the basis on which senior management can draw conclusions and identify follow-up actions with regard to the implementation of safeguards in a particular State. Within this framework the Environmental Sampling Programme constitutes an essential element to confirm that a State's declarations are correct and complete.

Environmental sampling for safeguards has been routinely implemented by the IAEA since 1996 and is one of the most important new technical measures introduced in recent years. IAEA inspectors take approximately 650 environmental swipe samples per year during routine inspections, design information verification and complementary access. The sensitive analysis of such samples enables the Agency to draw conclusions about past and present nuclear activities at the inspected location and is a powerful means to discover undeclared materials or activities in the presence of declared ones, or at undeclared locations.

Since 1996, the Agency has acquired a great deal of information about the analysis of environmental swipe samples at the Agency's Safeguards Analytical Laboratory (SAL) as well as at the other laboratories of the Network of Analytical Laboratories (NWAL) in Member States. Analysis methods such as high-resolution gamma spectrometry (HRGS), X-ray fluorescence spectrometry (XRF) thermal ionization mass spectrometry (TIMS) and secondary ion mass spectrometry (SIMS) are used to measure the uranium and plutonium in swipe samples to establish a fingerprint of the nuclear materials present and their processing history – enrichment, irradiation and chemical separation. These results are reported to the Environmental Sampling Database in the Department of Safeguards and are evaluated with statistical methods to arrive at conclusions about whether the materials found are consistent with the inspected State's declaration.

Currently employed analytical and evaluation methods are quite sophisticated; however, there are areas for improvement. Age dating methods and impurity measurements are needed to determine the processing history of the material on the swipes and clues to its origin. Detection of ever-smaller amounts of Pu is needed to improve the Agency's ability to detect a clandestine irradiation or reprocessing operation. In general, all analytical methods and the associated logistics (sample receipt, screening and dispatch) must be streamlined and improved to provide better results in a timely manner. Careful estimation of measurement uncertainty and detection limits by the measurement laboratories is crucial for ES evaluation, since analytical data are assessed using statistical methods and are compared to facility declarations and fuel cycle modelling calculations. In addition, the number of evaluations and the level of detail of such evaluations require that sufficient resources are available so as to enable the IAEA to provide high quality results in a timely fashion. The above issues will be examined in greater detail in this paper.

## 2. Analysis of Environmental Samples

Typically, up to 700 samples are taken and analysed per year. The sample kits are prepared in the Class-100 Clean Laboratory which is a part of the SAL. The samples normally contain six cotton swipes, two of which are analyzed by bulk and/or particle techniques, with the remaining portions being archived in SAL or, occasionally, stored by the plant operator/state authority for reference purposes. For the sampling in hot cells, special sampling kits have been developed. Furthermore, pre-inspection kits are used by inspectors to swipe themselves before entering the site to be inspected.

All non-radioactive environmental samples are received at the IAEA's Headquarters where they are coded to maintain confidentiality; radioactive samples are received and coded in a dedicated location in SAL. Moreover, after the coding every sample is screened in SAL using HRGS and/or XRF to measure the radioisotopes present and the U content. One of the advantages of performing the HRGS in SAL is that the measurement of short-lived isotopes can be carried out with minimum delay.

Based on the inspector's requirements and on the results of the screening, the Inspection Measurement Quality (IMQ) Unit of the Division of Safeguards Information Management prepares an analytical request. This request identifies the laboratory(ies) and the method(s) to be used for the analysis of the sample. It also takes into account the standing practice to obtain measurement results from two different laboratories and two different methods. In addition to the safeguards samples, quality control samples, blanks and known standards are sent to each member of the NWAL, where, bulk or particle measurements will be carried out. Therefore, measurements of environmental samples at SAL and the NWAL fall into three broad categories: screening, bulk analysis and particle analysis.

### 2.1. Screening

As mentioned before, all samples are screened at SAL for the presence of radioactive isotopes by HRGS and for the presence of U by XRF. These measurements are needed to guide the further handling (i.e. shipping) and analysis of the samples. The HRGS measurements also serve as a first detailed measurement of the fission product activities and is the most effective in detecting short-lived isotopes such as those coming from medical isotope production. By the time that samples arrive at a network laboratory many short-lived isotopes will have decayed to an activity below the detection limits of the receiving laboratory. HRGS is also capable of detecting the presence of U above about 5 micrograms and in addition (provided that sufficient counts are collected in the $^{235}U$ and $^{234}Th$ peaks) it

may be possible to estimate the enrichment of the U present (assuming equilibrium between $^{238}U$ and $^{234}Th$).

The equipment used for XRF screening for U is shown in Figure 1. It consists of a 100 W X-Ray tube with end window geometry, a robotic arm to position the sample and a Si(Li) detector for the fluorescent X-rays. The sample is a single swipe inside its plastic bag and held in a metal frame. The robotic arm positions the swipe in the measurement position approximately 2 cm from the X-ray tube and moves it in a "raster" motion to cover the entire surface (100 $cm^2$) in a series of measurements that takes approximately 4 hours per sample. A preliminary energy selection filter (PESF) based on Bragg scattering from graphite crystals is mounted on the Si(Li) detector to filter out unwanted X-ray quanta. The detection limit for U on a swipe is 35 $ng/cm^2$ and the final data format is a "map" showing where the U was detected on the swipe.

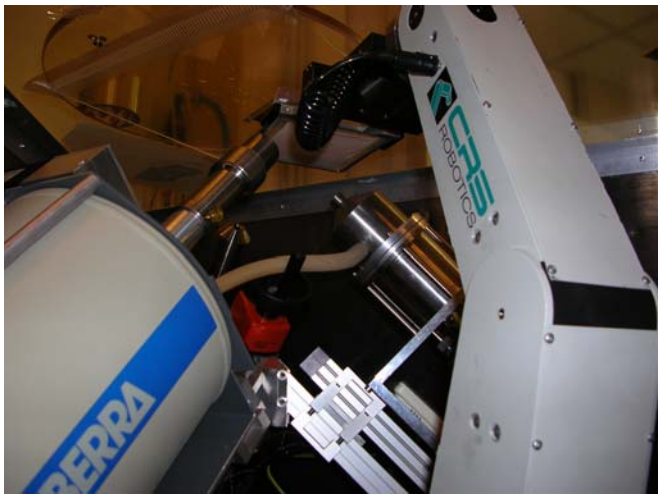


**Figure 1**. Inside view of XRF screening device showing robotic arm holding a swipe sample above the X-ray tube (right side) and Si(Li) detector with preliminary energy selection filter.

## 2.2 Bulk Analysis

The Department of Safeguards formulates about 200-400 analytical requests per year for ES bulk analysis. Apart from SAL, there are about seven laboratories in the NWAL that handle such requests albeit with different capacities, ranging from about 10 to 250 samples per year, and capabilities in terms of measurement uncertainties and detection limits.

Bulk IDMS is primarily used to measure the U and Pu quantity, the U isotopic composition ($^{233}U$, $^{234}U$, $^{235}U$, $^{236}U$, $^{238}U$), and the Pu isotopic composition ($^{238}Pu$, $^{239}Pu$, $^{240}Pu$, $^{241}Pu$, $^{242}Pu$). Currently, SAL performs 20% of all requested bulk analyses, whereas the remaining 80% of the bulk measurements are done in the NWAL.

For bulk analysis in SAL, the entire swipe is dissolved in mineral acids ($HNO_3$, HCl or $HClO_4$) following a preliminary ashing step in a furnace at $600^o$ C. The resulting solution is split to give an analytical portion and an archive portion which is used in case of problems. Chemical separation of U and Pu is accomplished with anion exchange chromatography and the use of stationary phase chromatography (Eichrom resins) to arrive at pure fractions of U or Pu for subsequent mass spectrometric analysis. The traditional method used in SAL is thermal ionization mass spectrometry (TIMS) using the instrument shown in Figure 2. The sample fractions are initially dried and then re-dissolved in a few microliters of nitric acid and dried on a Re filament to which is added high-purity graphite powder to enhance the ionization efficiency. Measurement of the isotopes of interest is accomplished with a single ion counting detector and peak jumping. The Triton TIMS instrument in SAL is also equipped with a multi-ion counting detector system, but this is not routinely used due to stability problems. Use of the MIC detectors would result in a factor of 2-3 higher efficiency of collection and improved counting statistics for the minor isotopes.

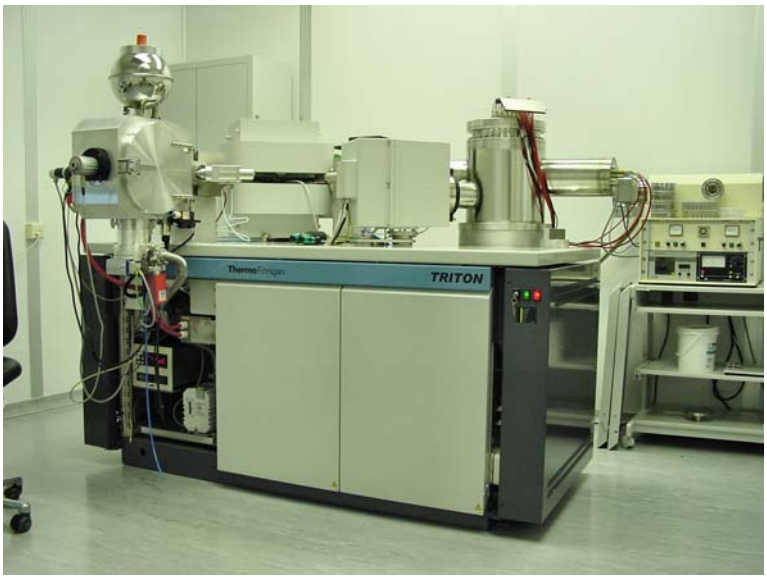


**Figure 2**. Thermal ionization mass spectrometer at SAL for bulk analysis.

The measured detection limits for TIMS at SAL are 70 fg of $^{239}$Pu and approximately 1 ng of natural U which is limited by the variability of the U concentration in the swipe material (Texwipe 304 cotton).

### 2.2.1 Future Improvements in Bulk Analysis

Although the need for extra capacity in the area of bulk analysis is not very urgent, an improvement in the capabilities would definitely be welcome. In this respect, it should be mentioned that several network laboratories have shifted from the use of TIMS to the implementation of inductively-coupled plasma mass spectrometry (ICP-MS) for the final measurement of U and Pu coming from bulk analysis of swipe samples. This is due to the high sensitivity of ICP-MS and freedom from interference effects coming from elements other than U or Pu which can affect the ionization efficiency of TIMS. ICP-MS therefore is more trouble-free and may require less stringent chemical separation procedures, thus leading to a more timely analysis. SAL is currently in the process of acquiring an ICP-MS (ELEMENT 2) to be placed in the Class-100 Clean Laboratory. It is expected that the detection limit for Pu will be less than 70 fg although the highest accuracy U isotope measurements will continue to be performed on the TIMS instrument. The MIC detectors on the TIMS instrument will continue to be investigated in order to address the short-term and long-term drift problems which have prevented their use in the past. Provided that these problems can be overcome, the use of MIC detectors will result in higher sensitivity and/or better accuracy for minor isotopes, i.e. $^{234}$U and $^{236}$U.

## 2.3 Particle Analysis

The IAEA normally issues 500-800 requests per year for particle analysis. From these about 40% are analysed by using SIMS and about 60% by using Fission Track-Thermal Ionization Mass Spectrometry (FT-TIMS). In case of SIMS, a dispersion of particles is loaded onto a carbon substrate, a “primary” ion beam such as $O_2^+$ is produced and used to sputter and ionize atoms coming from individual particles. These secondary ions are extracted, enter the mass spectrometer where they are separated according to mass and detected. The application of FT-TIMS requires that particles are irradiated in a thermal neutron flux in contact with a fission-track polymer (Lexan) which permits the particles containing fissile isotopes to be located under an optical microscope. The particles then are picked and loaded onto a filament, which is heated in the ion source of a TIMS instrument. The positive U or Pu ions which are created are extracted, and quantified in the mass spectrometer.

The SIMS is primarily used to measure the U isotopics, whereas FT-TIMS is used for determining both U- and Pu-isotopics.

The workhorse instrument at SAL for particle analysis has been the Cameca IMS-4f SIMS operating with a resistive anode encoder detector and PSEARCH software. Recent improvements have been obtained in the speed of analysis by use of a rapid field searching protocol to look for evidence of U particles in each field. Fields that do not contain a significant amount of U are measured for a few seconds only followed by switching to the next field. When U particles are found, the final measurement is performed in microprobe mode with a 1-2 micrometer primary ion beam and each particle is sputtered to exhaustion to obtain the most precise minor isotope data.

### 2.3.1 Improvements in Particle Analysis Methods

Normally, about 90 SIMS particle analyses are being carried out per year at SAL, whereas 200-300 are performed by other laboratories. There are about five network laboratories that provide the services of SIMS particle analyses. These laboratories offer various through-puts for IAEA samples, and with a range of analytical performance. Basically, all SIMS analyses are performed for U-isotopics.

The Pu-isotopics in particular are normally only measured by FT-TIMS, along with the U-isotopics. The uncertainties and detection limits are normally better than those obtained with SIMS. There are in principle only three members of the NWAL that perform FT-TIMS analysis for the IAEA. Due to the limited capacity of two of these FT-TIMS laboratories, about 95% of all the FT-TIMS particle analyses are performed by the third laboratory. Therefore, and in the light of expectations about the number of environmental samples and the analytical requirements, the Agency and its NWAL are exploring possibilities to install more FT-TIMS capabilities, or equivalent equipment, such as the ultra-high sensitivity SIMS (UHS-SIMS).

SAL staff have investigated the utility of UHS - SIMS measurements for safeguards purposes at the applications laboratory of Cameca and at the NORDSIMS laboratory in Stockholm. UHS-SIMS instrumentation (Cameca 1280) is physically bigger and more costly than standard SIMS (Cameca IMS-4f or 7f) but the transmission efficiency is 2-5 times higher and the mass resolution permits the rejection of common molecular interferences that give erroneous results for $^{235}U$ or $^{238}U$. Therefore, UHS-SIMS is the best commercial technology for obtaining high quality U isotopic data on micrometer-sized particles. A typical UHS-SIMS installation (at National Institute of Science and Technology, Gaithersburg, USA) is shown in Figure 3.

**Figure 3**. Cameca 1280 Ultra-High sensitivity SIMS instrument.

The requirement to measure micrometer-sized Pu particles has resulted in a development project at SAL to combine scanning electron microscopy (SEM) with chemical separation and ICP-MS measurements. The SEM is used to locate Pu-containing particles in the back-scattering image mode followed by energy-dispersive X-ray fluorescence spectrometry. Particles of interest are then measured with wavelength-dispersive XRF to estimate the Am/Pu and U/Pu ratios. These particles are

then relocated under an optical microscope and picked up with a micro-manipulator needle. The individual particles are subjected to chemical separation following a similar scheme to bulk analysis with the final measurement by ICP-MS to obtain the $^{240}Pu/^{239}Pu$ and $^{241}Pu/^{239}Pu$ ratios. The combination of SEM/XRF and ICP-MS data allows an estimate of the age of the Pu in each particle. This method is currently under development at SAL, but preliminary results on standard materials have demonstrated that the precision and accuracy of the method is acceptable.

## 3. Evaluation of Analytical Results

The analytical results produced by the laboratories are evaluated by the ES-Evaluation Unit in the Department of Safeguards. In total about 2000 analytical datasets (particle, bulk, HRGS, etc.) per year are evaluated for safeguards purposes. The evaluations are performed by, inter alia, using statistical tools such as “clustering” and fuel cycle modelling techniques, including various burn-up codes for issues related to irradiation and chemical separation, and centrifuge configuration codes for issues related to enrichment. In addition, blank and other standard samples are evaluated to monitor the performance of the respective laboratories. The results of the evaluations are reported to the relevant Divisions of Operations by grouping them per inspection. This means that up to 600 such reports are produced per year. Moreover, numerous summary reports on specific subjects are written upon request by the Division of Operations or by management. The evaluation results per state are summarized for inclusion in the respective SERs, which are issued annually. Currently, about 70 SER contributions are being produced per year.

This heavy workload is one of the two main causes of the delay in the production of the evaluation reports of samples that have routine priority. The other main cause is the time needed for analysis by the laboratories. The complete evaluation cycle for these kind of samples (from sample taking to final issue of an evaluation report and assessment of consistency) takes about 10 months, from which one month can be attributed to screening and shipment, four months to analysis, and five months to evaluation The delay is most apparent for the processing of particle samples. Fortunately, the total evaluation cycle for high priority samples can be completed in about three months. The number of high priority samples has increased over the past years, which causes extra delay in the processing of routine priority samples.

From the above it is clear that for streamlining and optimizing the environmental sampling programme there is a need to improve both the capacity and the capabilities of SAL and the NWAL, specifically in the area of particle analysis. This is being done through initiatives that aim to extend the analytical possibilities, inter alia, by increasing the number of laboratories participating in the programme, by adding powerful equipment such as UHS-SIMS, and by improving the performance of existing technologies. In addition, it is obvious that sufficient staffing with highly qualified evaluators is a pre-requisite for an optimal process.

The IAEA has also shown interest in new methodologies, e.g. in age dating of U and Pu on the swipes, and in determining impurities in the samples. These methodologies can be used to make some statements as to when the material was processed, and as to the processes and origin it stems from. Here again, the use of UHS-SIMS could be of high importance.

## 4. Conclusions

The analysis of environmental samples followed by the evaluation of results represents a powerful method to detect the presence of undeclared nuclear materials and activities at inspected locations. The IAEA’s ability to draw conclusions about the absence of such materials or activities rests on the sensitivity of both the analytical methods and evaluation tools used. The degree of confidence placed in such conclusions relies on the combination of highly sensitive and selective analysis methods, sophisticated evaluation methods, availability of evaluators and, perhaps most importantly, sample taking in the optimum locations to collect evidence of the most likely proliferation scenarios.

It is therefore important that the most sensitive analytical techniques and the best evaluation tools are explored by qualified people.

# Particle analysis for uranium isotopics on swipe samples using SIMS supported by SEM automated uranium detection

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## *Abstract:*

*This study examines the detection and the analysis for uranium isotopic composition in micrometer size particles extracted from swipes using scanning electron microscopy (SEM) and secondary ion mass spectrometry (SIMS) techniques. The use of low sputtering rates and high objective magnification allows sensitivity improvement of the SIMS analysis. Accurate 234/238 and 235/238 uranium isotopic ratios are determined in 1 µm diameter uranium oxide particles with relative precision of ± 6% and ± 0.7% (1σ), respectively.*

**Keywords:** SIMS; environmental samples; particle; uranium isotopes

## 1. Introduction

Recent safeguard issues revealed a need for increasing swipe samples analysis capability within the Agency's Network of Analytical Laboratories (NWAL). Our laboratory is qualified since 2001 to analyze uranium isotopics in particles using the Fission Tracks / Thermo-Ionization Mass Spectrometry method (FT-TIMS). In addition, we recently acquired a Secondary Ion Mass Spectrometer (SIMS) in order to broaden particle analysis capabilities in the laboratory. The SIMS technique enables to lower the response time for urgent analysis and to maintain a swipe sample analysis capacity even when our reactor for neutron irradiation is not available.

This presentation focuses on the results that have been obtained on particle analysis for uranium isotopes using our new generation Cameca IMS 7f. Particles are extracted from swipe samples. In this study, all sample processing steps have been examined with the view to improve the reliability of the results and to shorten analytical response time.

## 2. Particle extraction and sample mounting

All sample preparations are conducted in a class 10 clean room according to the method used by most laboratories that participate in the NWAL [1]. Particles are transferred from swipe to ethanol suspension by ultrasonification. The suspension was then evaporated to ~ 1.5 mL. Half volume was deposited onto a heated (40°C), diameter 25 mm carbon disk by means of a 100 µL micro-syringe allowing the output of 10-20 µL droplets. Finally, preparations are baked at 400°C for 2 hours to volatilize organic compounds from sample surface.

## 3. Uranium-bearing particle detection

The main difficulty in particle detection arises because sample screening software which are commonly used for SIMS automated uranium-bearing particle search [1] still have to be updated to a version compatible with the IMS 7f software (e.g. P-search by Evans Analytical).

### 3.1. SEM capability for automated detection of uranium-bearing particles

In this study, the automated detection of uranium-bearing particles is performed using a FEI XL 30 environmental SEM fitted with an EDAX system. An adaptation of the Gun Shot Residue forensic software allows the automatic search for uranium-containing particles using back-scattered electron image analysis and qualitative micro-analysis of major elemental composition by energy dispersed X-ray spectrometry. In addition, secondary electron images of uranium-containing particles can be acquired in order to characterize their morphology. An overnight GSR run may investigate a ~ 1 $cm^2$ deposition area, detecting with a high probability all uranium-bearing particles with diameter > 1 µm. The GSR program provides a listing of uranium-bearing particle coordinates relative to the SEM sample stage.

### 3.2. SEM to SIMS relocation of particles

Sample mountings are equipped with an internal reference consisting of 2 aluminum foil triangle pieces. This internal reference enables the determination of parameters in the transformation of coordinates relative to the SEM stage, to coordinates relative to the SIMS sample stage according to triangulation method [2] with a precision better than 50 µm. Then, sample surface is rastered with a 70 nA $O_2^+$ primary ion beam over a 400 µm×400 µm area centered on the calculated position of the particle of interest. In all the preparation analyzed, it appears that 100 % of the uranium-bearing particles previously detected by SEM could be pointed out on ion images acquired at mass 238 using microprobe mode.

Compared to manual SIMS detection, this method presents some advantages: the SEM/EDX detection is non-destructive (the whole particle is available for the IR measurement), non-susceptible to isobaric interferences, more efficient, faster, and provides some additional relevant information on individual particles (e.g. volume, morphology, and major elemental composition).

## 4. SIMS analysis of uranium isotopic ratios

### 4.1. Sensitivity improvement

Sensitivity is critical in particle analysis, considering the very small amounts of uranium to be analyzed (~ a few pg). Analyses of particles for isotopic ratios using monocollector small radius magnetic sector SIMS such as Cameca IMS7f instrument are commonly performed using high sensitivity instrumental configuration (i.e. low mass resolution, maximal energy pass band, large contrast diaphragm and field aperture) [1]. In this study, we investigate the influence of primary and secondary optical settings sensitivity during uranium isotopic analysis in synthetic particles of natural uranium oxide with diameter ~ 3 µm.

#### 4.1.2. Primary ion beam parameters

It is observed that sputtering for 1 hour by static primary ion beams (2, 5, and 20 nA intensity) does not allow reaching complete consumption of particles. According to recommendations in [4], one has thus to consider “practical sensitivity” as the appropriate measure of sensitivity (i.e. for uranium, the ratio between the number of detected secondary ions of uranium and the number of primary ions hitting the sample).

It appears that practical sensitivity is one order magnitude higher at low sputtering rate (primary ion intensity of 2 nA vs 20 nA). This result may indicate that low sputtering rate enhances secondary ionization rate of particle uranium.

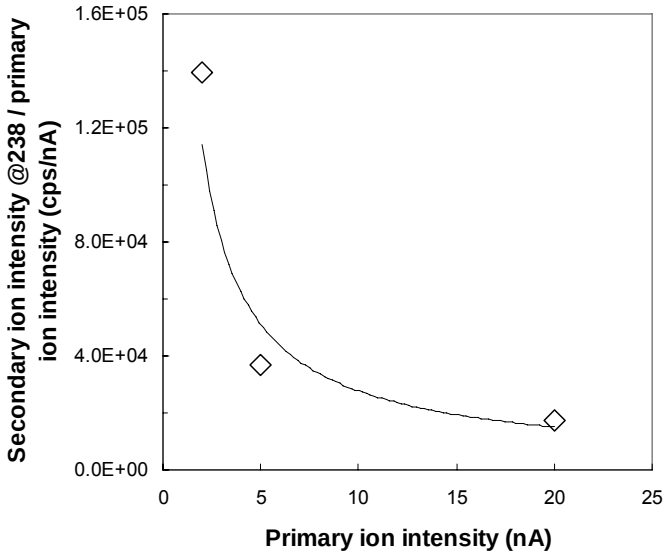


**Fig. 1:** Practical sensitivity as a function of primary ion intensity for 1 hour sputtering on 3 µm diameter uranium oxide particles. The value given for secondary ion intensity corresponds to the average over the experiment.

### 4.1.3. Transfer optics configuration

Besides, sensitivity improvement is obtained by improving transmission characteristics (i.e. the ratio between the number of ions detected and the number of ions formed). In this study, the configuration of transfer optics has been optimized according to [5].

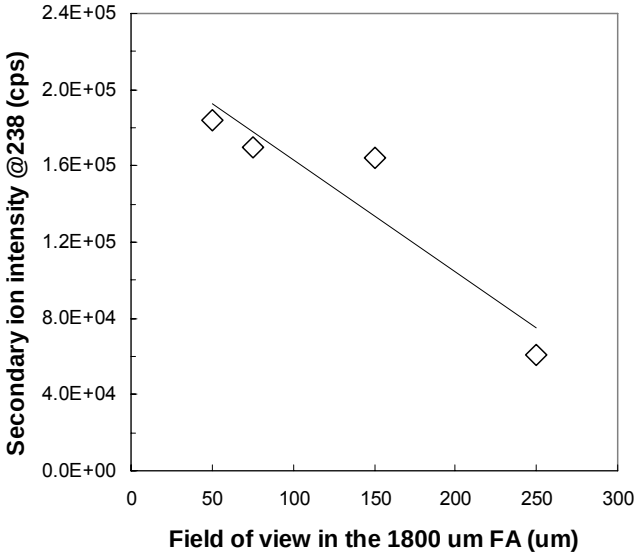


**Fig. 2:** Secondary 238-uranium ion intensity measured on 3 µm diameter uranium oxide particles as a function of transfer optics configuration. The values given on X-axis axis correspond to the diameter of the field of view fitting the 1800 µm field aperture (FA). Actually, this value is inversely proportional to ion image magnification.

The 2 transfer electrostatic lenses are set such as to maximize the magnitude of the ion image at the plane of the field aperture. This has the effect of reducing ion beam crossover diameter and thus decreasing the part of secondary ions crosscut at the entrance slit and contrast diaphragm plane.

#### 4.1.4 Spectrometer settings and optimization of mass peak shape

Besides, this study points out that analyzer setting is a key point for high sensitivity analysis. The trajectories that do not cross the field aperture and the entrance slit at their respective center will not be focused at the exit slit plane. The use of large field aperture (1800 µm) and large energy slit (125 eV) during high sensitivity analyses increase the blurring of the entrance slit image at the exit slit plane due to second order aberrations. When the coupling between the spectrometer lens and the electrostatic sector is not optimized, we commonly observed that the transmission to the analyzer of ions with > 100 eV energy range leads to strong asymmetry of aberrations at the exit slit plane. This aberration feature appears to be correlated with dissymmetric mass peak geometry which is not suitable for the determination of precise isotopic ratios.

### 4.2. Repeatability, reproducibility, and accuracy of measurements

In order to evaluate the reliability of the measurements, we acquired synthetic particle samples with diameter calibrated to ~1 µm that were prepared at ITU from NIST certified reference materials (CRMs): U-010, U-020, U-030, and U-500 [3].

#### 4.2.1 Repeatability and reproducibility

We use primary ion beam intensity of 3 nA and transfer optics of 50 µm, corresponding to higher practical sensitivity. Typical secondary uranium ion intensities are for instance in U-010 sample: $3\times10^5$ cps, $3\times10^3$ cps, 20 cps at mass 238, 235, and 234, respectively. Counting times are set at 0.5 s, 1 s, and 2 s, respectively. An analysis run is divided in 12 cycles. Under this conditions, a repeatability (i.e. the relative standard deviation of the mean over the isotopic ratios measured during an analysis run) ranging between 0.5% and 1.5 % and between 2% and 8% is obtained on $^{235}U/^{238}U$ and $^{234}U/^{238}U$ ratios, respectively. This value is in rather good agreement with the theorical relative standard deviation of the mean calculated from the Poisson's law (between 0.3% and 0.8% and between 3% and 7% on $^{235}U/^{238}U$ and $^{234}U/^{238}U$ ratios, respectively).

Point-to-point reproducibility is estimated from the standard deviation over isotopic ratios measured over different particles (Fig. 3, Table 1). The values obtained are similar to the within-run repeatability measured on the 4 samples. This indicates that the analytical conditions can be considered as reproducible from particle to particle in the synthetic samples.

#### 4.2.2. accuracy

Moreover, a good agreement is also observed between SIMS isotopic measurements and certified values for $^{234}U/^{238}U$ and $^{235}U/^{238}U$ (table 1). Average instrumental mass fractionation factors (ie. the ratio between measured and certified isotopic ratios) over the 4 samples are 1.0083 ± 0.0052 and 1.036 ± 0.021 for $^{235}U/^{238}U$ and $^{234}U/^{238}U$ ratios, respectively. This means that a correction for instrumental mass fractionation has to be applied to SIMS measurements in order to obtain accurate results. It should be noted that this correction is an additional source of uncertainty of the final results. Error propagation calculations give typical relative uncertainty (1σ) on corrected SIMS measurements ranging between ± 0.5% and ± 1.5% (average: ± 0.7%) and between ± 3% and ± 13% (average: ± 6%), for $^{235}U/^{238}U$ and $^{234}U/^{238}U$ ratios, respectively.

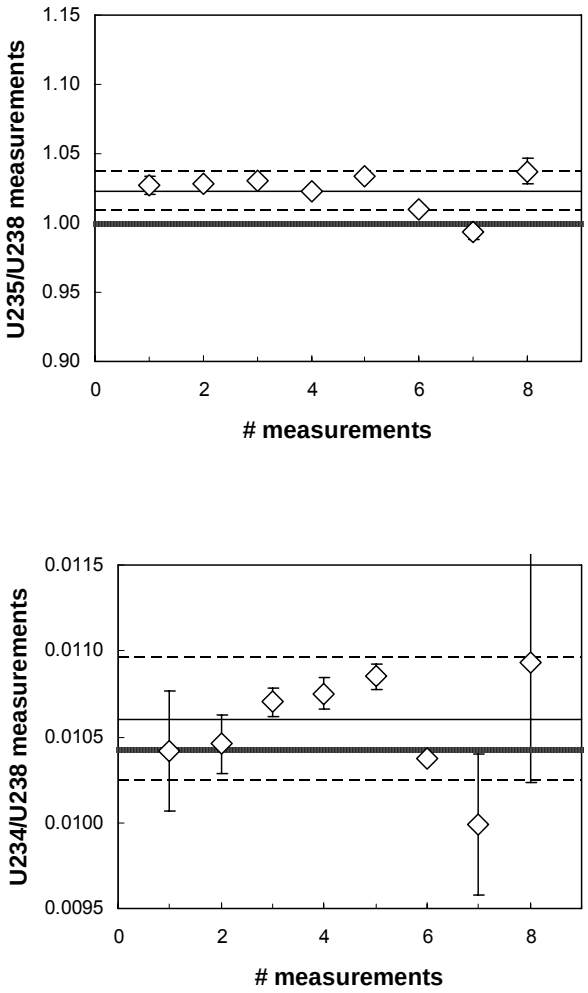


**Fig. 3:** Uranium isotopic ratios measured on 8 different 1 µm diameter particles of NIST certified reference materials (CRM) U-500. All the analysis are made using the same analytical settings. Error bars correspond to the within run repeatability. Solid and dashed lines indicate the mean and the standard deviation over the 8 measurements, respectively. The bold lines represent the certified values for 235/238 and 234/238 uranium isotopic ratios.

| | U-010 | U-020 | U-030 | U-500 |
|---|---|---|---|---|
| $^{235}U/^{238}U$ measurements[a] | (1.0145 ± 0.0050) $\times 10^{-2}$ | (2.0723 ± 0.016) $\times 10^{-2}$ | (3.167 ± 0.049) $\times 10^{-2}$ | 1.0230 ± 0.0037 |
| Certified Value | (1.01400 ± 0.00051) $\times 10^{-2}$ | (2.06872 ± 0.00057)$\times 10^{-2}$ | (3.142960 ± 0.00059)$\times 10^{-2}$ | (9.997 ± 0.010)$\times 10^{-1}$ |
| Relative Std Error[b] (1σ, within run) | 0.50% | 0.76% | 1.6% | 0.36% |
| Reproducibility[c] (1σ, point-to-point) | ± 0.0072 $\times 10^{-2}$ | ± 0.0098 $\times 10^{-2}$ | ± 0.089 $\times 10^{-2}$ | ± 0.014 |
| Number of analyzed particles | n=7 | n=6 | n=9 | n=8 |
| Average IMF factor[d] | 1.0005 ± 0.0050 | 1.0020 ± 0.0076 | 1.008 ± 0.016 | 1.0233 ± 0.0039 |

| | U-010 | U-020 | U-030 | U-500 |
|---|---|---|---|---|
| $^{234}U/^{238}U$ measurements[a] | $(5.71 \pm 0.65) \times 10^{-5}$ | $(1.93 \pm 0.18) \times 10^{-4}$ | $(1.14 \pm 0.24) \times 10^{-4}$ | $(1.060 \pm 0.022) \times 10^{-2}$ |
| Certified Value (1 σ) | $(5.465 \pm 0.025) \times 10^{-5}$ | $(1.7683 \pm 0.0015) \times 10^{-4}$ | $(1.9605 \pm 0.0016) \times 10^{-4}$ | $(1.0422 \pm 0.0013) \times 10^{-2}$ |
| Relative Std Error[b] (1σ, within run) | ± 5.5% | ± 8.2% | ± 13% | ± 2.2% |
| Reproducibility[c] (1σ, point-to-point) | $\pm 0.65 \times 10^{-5}$ | $\pm 0.32 \times 10^{-4}$ | $\pm 0.32 \times 10^{-4}$ | $\pm 0.036 \times 10^{-2}$ |
| Number of analyzed particles | n=7 | n=6 | n=9 | n=8 |
| Average IMF factor[d] | 1.046 ± 0.057 | 1.09 ± 0.10 | 9.92 ± 0.058 | 1.017 ± 0.021 |

[a] The uncertainty corresponds to the standard deviation of the mean over measurement cycles (i.e. within-run repeatability).
[b] Relative standard deviation of the mean over measurement cycles (i.e. within-run relative repeatability).
[c] Standard deviation over measurements on n different particles (i.e. point-to-point reproducibility).
[d] Instrumental Mass Fractionation (IMF) equals to the ratio between measured and certified isotopic ratios. The value given corresponds to the mean ratio over the n measured particles. The associated uncertainty corresponds to the standard deviation over the n measurements.

**Table 1:** Results obtained on 1 µm diameter particles of NIST certified reference materials (CRM) U-010, U-020, U-030, U-500.

## 5. Conclusions

The combination of SEM and SIMS techniques allows the detection of uranium-bearing particles overnight and the analysis for isotopic ratios of 40 particles daily. SIMS technique allows the determination of accurate 235/238 and 234/238 uranium isotopic ratios on 1µm diameter particles with a relative precision (1σ) of about ± 0.7% and 6%, respectively. The main drawback of this technique with regards to fission tracks / TIMS method is that it is not sensitive to 235U-enrichment of the detected particles. As a consequence, no priority can be drawn among the particles to be analyzed for isotopic ratios.

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# Direct alpha spectrometry from swipe samples

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***Abstract:***

*The use of direct high-resolution alpha spectrometry for swipe samples is demonstrated by measurements and simulations. A test sample was obtained from a natural uranium source using an IAEA kit for swipe samples. The swipe is placed as such into a vacuum chamber of a high-resolution alpha spectrometer, i.e. without any radiochemical sample manipulation. Monte Carlo simulations are used for characterizing those source material properties that influence the shape of the alpha peaks. The measured alpha particle energy spectrum is compared with the simulated energy spectra and an excellent agreement is found. Simulations suggest that spectra of better quality may be obtained if the sampling material is selected according to its suitability for direct alpha spectrometry. A new sampling method, based on surface stimulation and collection of radioactive particles emitted from the surface, is proposed. The method is tested using naturally occurring $^{210}$Po as a marker nuclide. It is concluded that direct alpha spectrometry can be used as a screening method for those samples that will be analyzed later by other means.*

**Keywords:** direct alpha spectrometry; swipe sample; sample screening; Monte Carlo simulation

## 1. Introduction

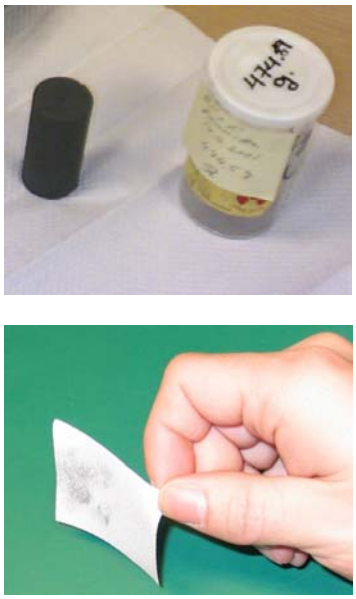
Sampling of radioactive material from surfaces by swiping is a simple method that can be applied even in harsh environmental conditions. The swipes can be readily analyzed using non-destructive methods, such as gamma-ray spectrometry, which facilitates sample screening for subsequent analyses. However, gamma-ray spectrometry alone is not necessarily the best option because most alpha-particle emitting radionuclides cannot be easily detected owing to their low gamma-ray yield. For these nuclides direct alpha spectrometry from the swipes should be considered.

In direct alpha spectrometry the alpha particles emitted by the source are counted without radiochemical sample manipulation. The method is non-destructive, which means that the samples are available as such for other analysis methods. Since there is no tedious radiochemical sample treatment the method is also cheap. The disadvantage is that only thin samples can be used in the data acquisition. Otherwise, the spectra may be complicated due to the alpha particle energy absorption in the source matrix. Thus, the key issue is to develop the sampling method more appropriate for direct alpha spectrometry

## 2. Direct alpha spectrometry

In alpha spectrometry the source characteristics have a major influence on the measured alpha particle energy spectrum. Most important source properties are the size of the particles containing radioactive material and particle penetration into the swipe material. For demonstration, a sample from oxidized surface of a natural U piece was taken using the IAEA kit for swipe samples. The sample was placed as such in a vacuum chamber of an alpha spectrometer equipped with a high-resolution PIPS detector.

The spectrum presented in Figure 1 was obtained using source to detector distance of 13 mm and data acquisition time of 10 h. The main peaks of $^{238}$U ($E_\alpha$ = 4.2 MeV) and $^{234}$U ($E_\alpha$ = 4.7 MeV) can be easily distinguished from the spectrum. However, $^{235}$U ($E_\alpha$ = 4.4 MeV) cannot be identified. Should the sample contain other fissionable material emitting alpha particles of higher energy, such as $^{239}$Pu ($E_\alpha$ = 5.2 MeV), it would very likely be detected as well.



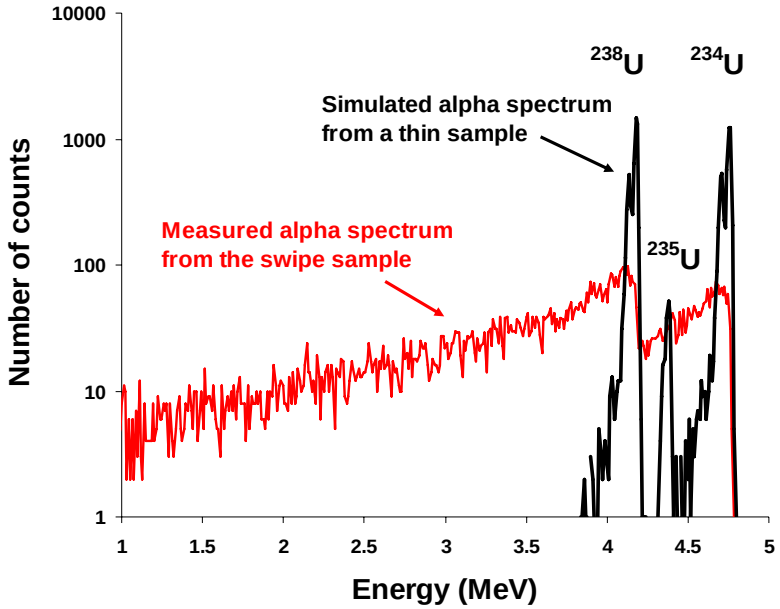



**Figure 1:** A piece of natural U (black object in the left), a swipe sample taken from it and measured alpha particle energy spectrum from the swipe. Simulated energy spectrum represents a thin (< 0.1 µm) sample.

## 3. Monte Carlo simulations of alpha spectra

On the basis of Figure 1 it is evident that direct alpha spectrometry for traditional swipe samples is not necessary the best option for sample screening. However, selecting the appropriate swipe material and the swiping method itself may drastically affect the quality (in terms of the peak width) of the alpha spectra. Monte Carlo code known as AASI [1] was used to examine these issues. AASI was developed for investigating those source properties that influence the α-peak shape, for generating synthetic alpha spectra and for training and learning purposes. The code takes into account properties of the detector and various characteristics of the source and the material between the source and the detector.

As shown in the previous publications (see e.g. [2-3]) simulated spectra are in agreement with those obtained from measurements. This is evident also in this case (Figure 2 A). Simulations can be used for investigating the effects of free parameter values, such as particle size and density (Table 1). The effects of the U particle size, the proportion of the particles penetrated the swipe material and the mean penetration depth are illustrated in Figure 2 B–D. The larger the particle size, the more round-edged the peaks in the alpha spectra, and the deeper the penetration depth, the wider the tails of the alpha peaks. It was also seen that the higher the proportion of the particles that penetrated the swipe, the higher the number of counts in the peak tails.

| | |
|---|---|
| Nuclides taken into account in the simulations | $^{234}$U, $^{235}$U, $^{238}$U |
| Particle composition | $UO_2$ |
| Particle density | 10.96 g/cm$^3$ (nominal density of $UO_2$) |
| Particle diameter | 0.6 µm (0.1 µm and 3 µm in Fig. 2B) |
| Mean penetration depth | 50 µm (20 µm and 5 µm in Fig. 2C) |
| Proportion of particles penetrated the swipe material | 85% (20% and 5% in Fig. 2D) |
| Swipe material composition | Si |
| Swipe material thickness and density | 0.5 mm, 0.3 g/cm$^3$ |

**Table 1:** Parameter values used in the simulations (see Figure 2A).

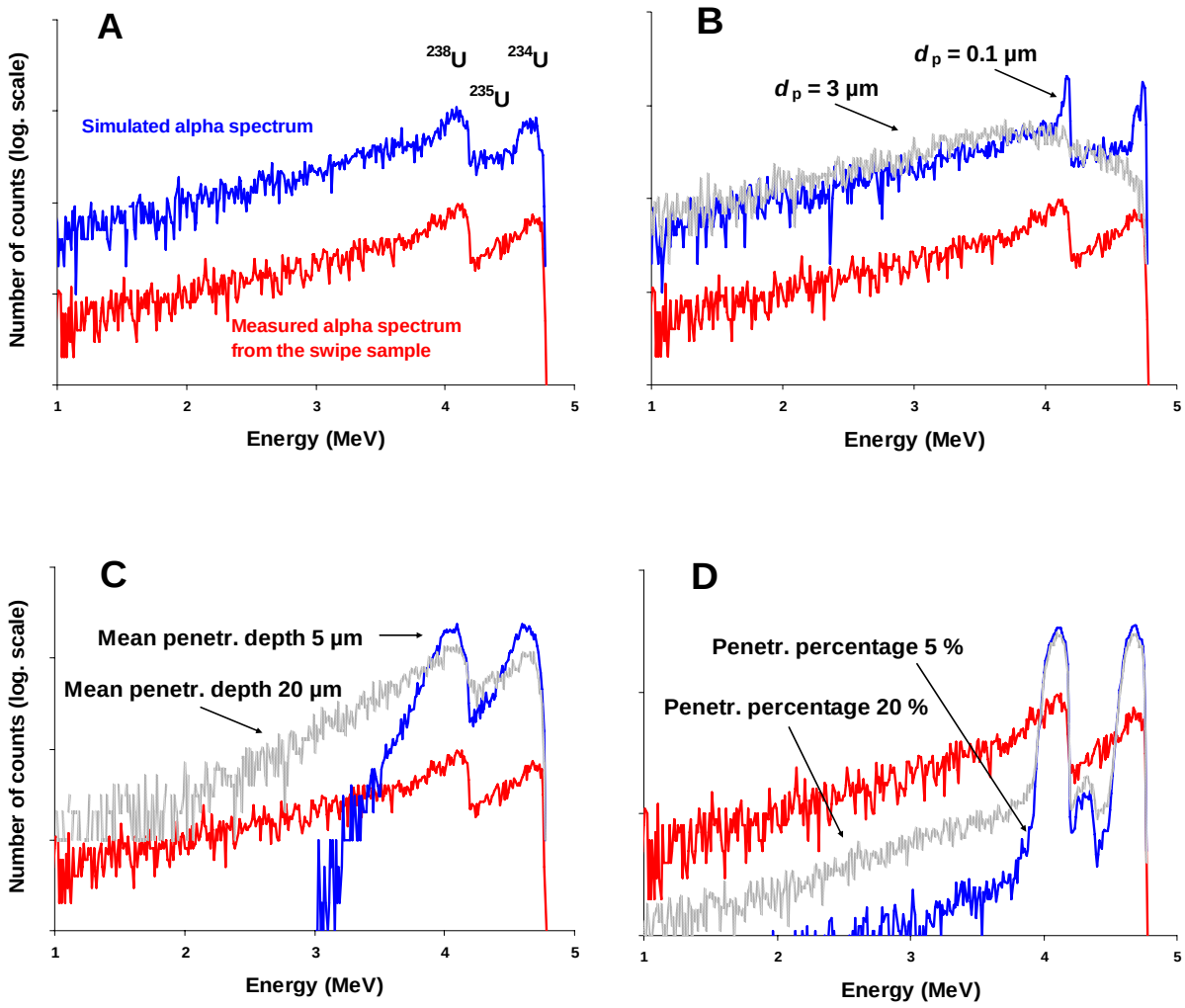



**Figure 2:** Alpha particle energy spectra obtained from the measurement (same spectrum present in A–D) and from simulations. Logarithmic y-axis is used for all figures. A) Parameter values presented in Table 1 are used in the simulations. The number of counts in the simulated spectrum is by a factor of 10 higher than in the measured one to distinguish the spectra. B) The effect of particle size. C) The effect of particle mean penetration depth into the swipe material. D) The effect of the proportion of particles penetrated the swipe. Other particles are assumed to remain on the sample surface.

## 4. A novel sampling method from surfaces containing radioactive material

It is shown above that the properties of the swipe material have an essential influence on the quality of the alpha peaks. In addition, characteristics of the sampled particles may also affect the spectra notably. Thus, selecting the sample material properly and using a more sophisticated sampling method, alpha particle energy spectra of good quality may be obtained. The peaks are not so sharp than in case of radiochemically manipulated sources but nuclide identification may be possible provided that the nominal energies of alpha particles are larger than about 50 keV. Thus, the sample screening might be possible using direct alpha spectrometry.

Unfortunately, for the time being, no well-tested substitutive method for swipe sampling exists. The new method should be as simple and robust as possible to facilitate flexible operation in field conditions. In addition, the equipment should be lightweight, compact and independent on electricity. Technical implementation of the proposed equipment consists of the following parts:

- Stimulator of the surface from which the particle samples will be taken.
- A commercial hand-held battery-driven air sampler which collects particles ejected from the surface.
- A nozzle containing appropriate air filter and a sampling tube to the sampler.

Purpose of the stimulator is to eject the radioactive particles from the surface. The stimulator must be a hand-held and easy-to-use tool that shoots high-pressure air, ultrasound, photons (laser), etc. to the surface resulting removal of the particles. A simple stimulator could be, for example, a bottle of an aerosol spray containing pure air. This kind of spray is often used for removing dust from sensitive electrical devices. Design and implementation of a proper stimulator is a key issue that should be further investigated to find an optimal solution for different types of surfaces.

Dispersion of the radioactive particles removed from the surface to the surrounding air is prevented using a specially designed nozzle connected to the sampling tube of the sampler. The tube is similar to those used in standard vacuum cleaners. The thin-walled nozzle is equipped with an aerosol filter that can be easily changed. Alternatively, the entire nozzle (or part of it) containing the filter can be disposable, which facilitates further manipulation of the samples and avoids cross contamination. During the sampling, the nozzle will be placed just above the surface (the walls are in contact with the surface) from which the radioactive particles are released. Typical flow rate of a portable accumulator-driven sampler is of the order of 10 $m^3$/h (≈ 3 l/s) depending on the filter type; typical operation time may be several hours depending on the flow rate.

To test the suggested method, a sample was taken in a garage from the surface of a brick wall. No additional stimulator for the surface was used, i.e. the particles were detached from the surface only by the air flow and low pressure generated by the sampler pump. Fluoropore membrane filter was used for collecting the released particles. Radon concentration in the garage air is typically 100 Bq/$m^3$. Natural radionuclides that can be detected in the air using direct alpha spectrometry are $^{218}$Po, $^{214}$Po (U-series) and $^{212}$Po, $^{212}$Bi (Th-series); $^{210}$Po cannot be detected because of its long half life ($t_{½}$ = 138 d) and short residence time in the air. However, $^{210}$Po serves as an indicator for testing the proposed sampling method: the method is viable if $^{210}$Po can be identified in the particles detached from the surface (Figure 3). This is because $^{210}$Po is generated by the decay of its long-lived precursor $^{210}$Pb ($t_{½}$ = 22.3 a) in the brick wall surface.

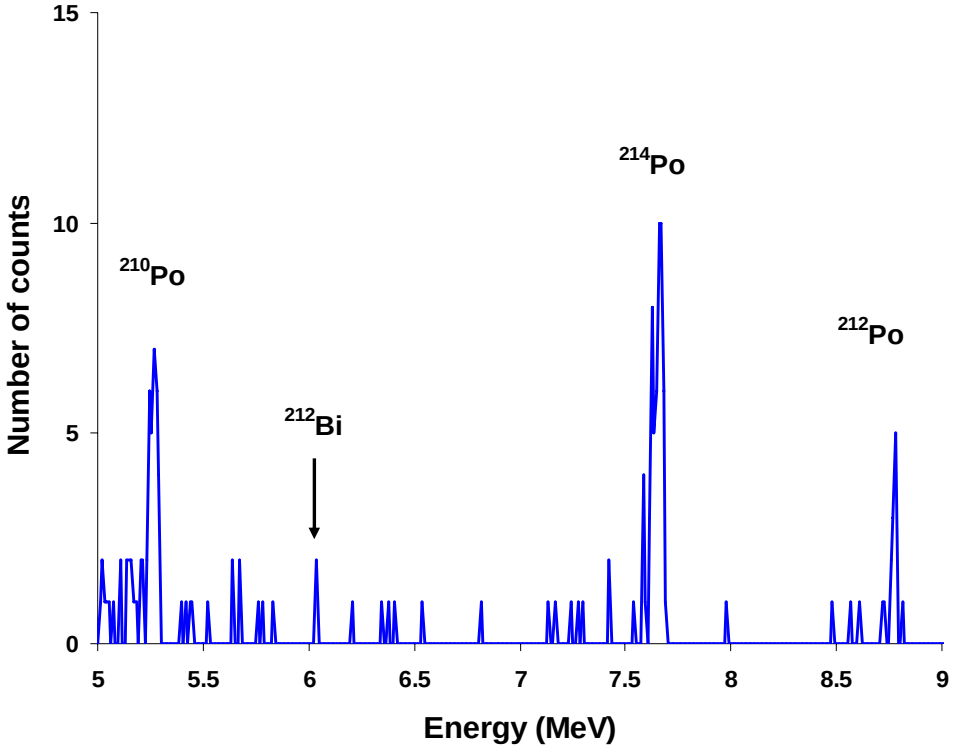



**Figure 3:** Alpha spectrum of the particles detached from the brick wall. $^{212}$Bi, $^{214}$Po and $^{212}$Po are typical naturally occurring airborne radionuclides whereas the long-lived $^{210}$Po is generated on the brick wall surface through the decay of $^{210}$Pb. In the measurement, the source to detector distance was 10 mm and source diameter was 27 mm which gives for the geometrical detection efficiency $\varepsilon$ = 0.13 [1]. Data acquisition time was 3 d which gives for the activity of $^{210}$Po in the air filter 1.7 mBq. Surface area from which the particles were detached was 5 $cm^2$ and, thus, the surface activity of $^{210}$Po was 0.34 mBq/$cm^2$. This value represents a lower limit because the removal efficiency of the particles from the surface is not known.

Simplified decay schemes of radon ($^{222}$Rn) and thoron ($^{220}$Rn) progeny are as follows (alpha-particle emitting airborne nuclides that are usually present in air filters are marked as bold):

U-series: $^{222}\text{Rn} \xrightarrow{\alpha} \mathbf{^{218}Po} \xrightarrow{\alpha} {}^{214}\text{Pb} \xrightarrow{\beta} {}^{214}\text{Bi} \xrightarrow{\beta} \mathbf{^{214}Po} \xrightarrow{\alpha} {}^{210}\text{Pb} \xrightarrow{\beta} {}^{210}\text{Bi} \xrightarrow{\beta} {}^{210}\underline{\underline{\text{Po}}} \xrightarrow{\alpha} {}^{206}\text{Pb}$ (stable)

Th-series: $^{220}\text{Rn} \xrightarrow{\alpha} {}^{216}\text{Po} \xrightarrow{\alpha} {}^{212}\text{Pb} \xrightarrow{\beta} \mathbf{^{212}Bi} \xrightarrow{\beta} \mathbf{^{212}Po} \xrightarrow{\alpha} {}^{208}\text{Pb}$ (stable)

The presence of $^{210}$Po is identified in the surface particle sample (Figure 3) which is an indication of the practicality of the suggested method.

## 5. Analysis of alpha spectra

Spectrum analysis in direct alpha spectrometry is more complicated compared to alpha spectrometry with radiochemical sample manipulation because of the alpha particle energy absorption in the source material and the presence of multiple overlapping peaks in the spectrum. Deconvolution of complex alpha spectra requires novel spectrum analysis tools, such as those proposed by Pöllänen and Siiskonen [4]. Development of such a code system is in progress. The basic idea is that the peak shapes used in the fitting are obtained through simulations (Figure 3), not using predefined functions.

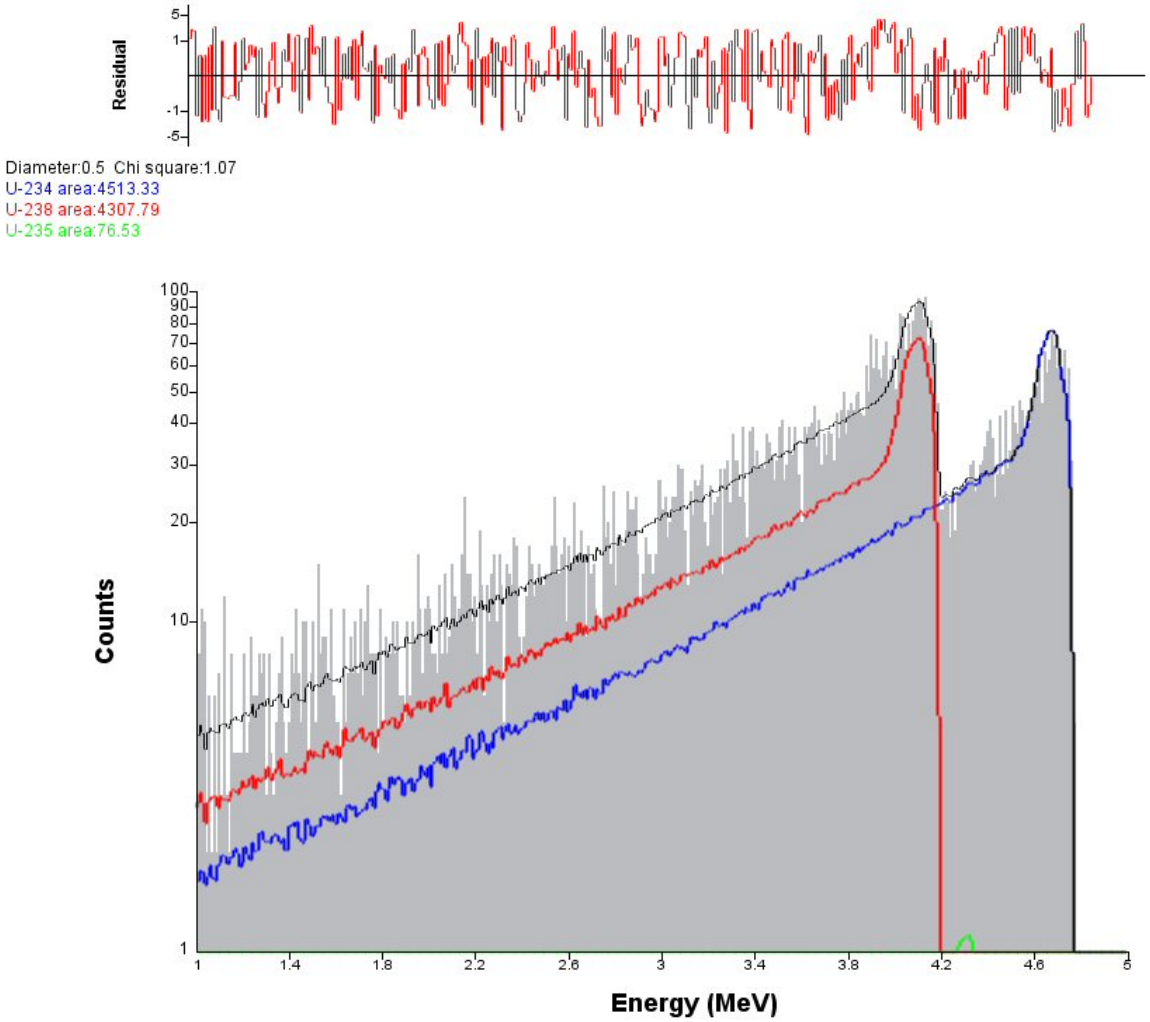



**Figure 3:** Unfolding the measured alpha particle energy spectrum (same spectrum as in Figure 1, grey area) using the AASI computer code. Residual, i.e. difference between the measured spectrum and fitted peak shapes, is presented above.

## 6. Conclusions

Compared to traditional swiping the suggested sampling method is more sophisticated. In swiping no other equipment than bare hands and the swiping material itself are needed. Here, additional equipment is inevitable to ensure that particles can be removed from the surface and, subsequently, all of them are collected in an air filter. The obvious advantage with this method is that the samples are readily available as such (i.e. no sample manipulation, such as sample cutting and division, is necessary) for different non-destructive analysis/screening methods. In addition, the sampling is optimized to facilitate the use of direct alpha spectrometry.

Thus, the proposed method aims at shortening and simplifying the chain of sampling, sample screening and sample analysis. This facilitates reorientation of the resources in a laboratory since, especially in the sample screening, tedious and expensive radiochemical sample manipulation may be replaced by spectrum analysis. Although the spectra may be more complicated than those obtained by radioelement separation, novel analysis codes may be applied for spectrum deconvolution.

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# Fluorine as a safeguards tool for age dating of uranium oxyfluoride particles?

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***Abstract:***

*When uranium hexafluoride ($UF_6$) is released into the environment, it reacts with the atmospheric moisture forming uranium oxyfluoride particles and HF. Fluorine-containing compounds such as HF are recognized as signatures for enrichment activities, but in this study the connection between the fluorine in the particles and the age of the particles has been investigated. Establishing this link is not evident, as uranium oxyfluoride particles are highly hygroscopic and little is known about their reaction mechanisms and stability. There are various environmental factors such as time, humidity and light or heat exposure that could affect the amount of fluorine, and even after the particles are collected the fluorine could decrease further due to sample preparation or storage conditions.*
*The influence of (ultraviolet) light exposure combined with the time elapsed after preparation has been investigated on uranium oxyfluoride particles prepared at IRMM using the aerosol deposition chamber. The presence of fluorine in single particles was confirmed by SEM-EDX. Semi-quantitative information was obtained by evaluating specific SIMS peak height ratios. With these measurements, possible correlations existing between the amount of fluorine and the age of a particle could be investigated, taking into account the environmental conditions to which the uranium oxyfluoride particles were exposed.*

**Keywords:** safeguards; uranium hexafluoride; uranium oxyfluoride particles; SEM-EDX; SIMS

## 1. Introduction

Environmental sampling introduced in the 1990's as part of the Additional Protocol of the IAEA, has proven to be an important safeguards tool for the verification of the absence of undeclared nuclear activities [1, 2]. By wiping surfaces in or around nuclear facilities using small pieces of cotton cloth called swipes, uranium-bearing particles are collected among millions of other dust particles. The analysis of these uranium particles can reveal key information on the site's current and past activities. Although the emphasis is still on the analysis of the uranium isotopes, a lot of information can be deducted from the particle's morphology, surface structure, crystallinity or elemental composition [3].
In the specific case of enrichment facilities, uranium oxyfluoride ($UO_2F_2$) particles are formed when uranium hexafluoride ($UF_6$) reacts with moisture from the atmosphere. As a result, particles formed from $UF_6$ of uranium enrichment facilities typically contain fluorine. In this study, we investigated whether the analysis of fluorine in oxyfluoride particles could complement the information on the uranium isotopic composition. What is more, if a correlation can be found between the amount of fluorine and the age of a particle, the analysis of fluorine in particles will not only provide information on the source of the particles, i.e. enrichment activities, but also on their history. However, there are many variables during and after particle formation that could affect the amount of fluorine, including humidity, light and heat exposure and the time elapsed after formation. As a result, the study of the analysis of fluorine in $UO_2F_2$ particles requires a careful design to take into account the environmental parameters that could have influenced the amount of fluorine. In this work, the effect of ultraviolet light and the time elapsed after preparation were examined in uranium oxyfluoride particles prepared by the aerosol deposition chamber at the Institute for Reference Materials and Measurements (EC-JRC-

IRMM). The influence of ultraviolet light and storage time were assessed by evaluating specific secondary ion mass spectrometry (SIMS) peak height ratios with the instrument in the ion microprobe mode. A scanning electron microscope equipped with an energy dispersive X-ray detector (SEM-EDX) was used to locate the uranium particles and to determine their elemental composition.

## 2. Materials and Methods

The relative amount of fluorine was evaluated in uranium particles produced in the aerosol deposition chamber. Uranium oxyfluoride particles are formed through the controlled hydrolysis of $UF_6$: when released into the deposition chamber, the $UF_6$ reacts with the atmospheric moisture forming HF and uranium oxyfluoride in the form of particles [4]:

$$UF_6 + H_2O \rightarrow UO_2F_2 + 4HF$$

The uranium oxyfluoride particles settle on graphite planchets of 25 mm diameter (Schunk, Germany) at the base of the chamber. These planchets can be directly inserted into the various measurement instruments including SEM-EDX and SIMS.
It has been demonstrated that the humidity of the air inside the deposition chamber is a determining factor for the particle morphology [4, 5]. A dry atmosphere of less than 15 % relative humidity, leads to submicron particles which agglomerate to larger structures of up to 100 micrometers. On the other hand, when the relative humidity of the air exceeds 60 %, the particles become more spherical and the degree of agglomeration is lower. These particles are separated by at least a few micrometers and are therefore better suited for SIMS analysis on individual particles. The diameter of these particles varies between a few hundred nanometers to 2.5 micrometers (Fig. 1).

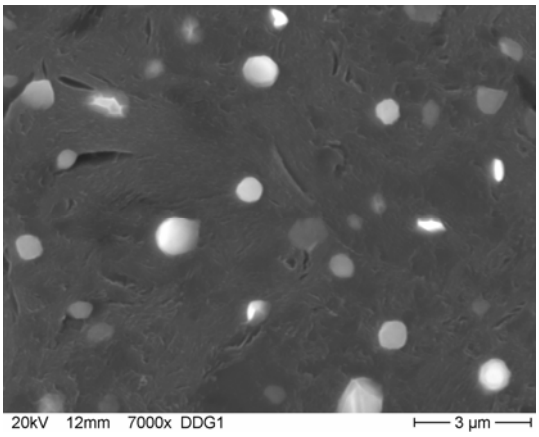


**Figure 1:** Scanning electron image in secondary electron mode showing uranium particles (white dots) produced in high humidity conditions (> 60 %) on a graphite planchet.

The particle morphology and composition were checked by a JEOL 6310 SEM-EDX instrument of the Belgian Nuclear Research Centre (SCK•CEN, Mol, Belgium) and at Antwerp University using a JEOL 6300 SEM.
At QinetiQ in Malvern (UK), a Cameca 4f spectrometer was used with 8.5 keV $O_2^+$ primary ion bombardment at a current of 2 nA in a focused spot (estimated diameter 10 µm).
SIMS ion microprobe measurements were applied to individual particles. The data were obtained by cycling the masses 238 (U), 239 (UH), 257 (UF), 273 (UOF) and 276 ($UF_2$) 10 times each to reveal any ratio variations with sputtering.

## 3. Results

The particles formed from uranium hexafluoride are highly hygroscopic and little is known about their long-term stability [4, 6]. Before being collected on swipes they might have been exposed to high temperatures, a high humidity or sunlight. All these environmental factors could have altered their morphology and composition, and more specifically, their fluorine content.
To study these effects, uranium oxyfluoride particles were prepared from uranium hexafluoride in the aerosol deposition chamber developed at IRMM.
The signature of freshly prepared particles was recorded to serve as a reference level. The particle composition was evaluated as explained above using SEM-EDX and SIMS.

As part of the original preparation procedure, the particles were heated for at least 6 hours in an open furnace at 350 ºC to remove excess water and other volatile elements. This heat-treatment inevitably removes most of the fluorine in the particles and as a consequence any existing information between the amount of fluorine and the age of a particle is lost.
The heat-treatment was therefore omitted from the preparation procedure and freshly prepared particles were measured directly by SEM-EDX after preparation. In contrast to the EDX spectra of the annealed particles, the spectra of these particles clearly showed the fluorine $K_\alpha$ line at 0.677 keV addition to the U $M_\alpha$ (3.17 keV) and $M_\beta$ (3.34 keV) lines.

SIMS ion-microprobe measurements were carried out to assess the effect of various exposure and storage conditions on the relative amount of fluorine in single particles. The relative strengths of fluorine-containing peaks in SIMS mass spectra were evaluated in particle samples each having different storage characteristics. All but one sample contained $UO_2F_2$ particles: one sample contained $UF_4$ particles stored in a laboratory environment for more than 2 years and was used to test the repeatability of SIMS $UF_4$ spectra and their quantitative distinctiveness from $UO_2F_2$. The storage time for the $UO_2F_2$ particle samples varied between 2 weeks to 29 months. One sample was UV-exposed for 3 weeks. For every sample, SIMS ion-microprobe measurements were applied to between 4 and 7 particles. The data were obtained by cycling the masses 238 (U), 239 (UH), 257 (UF), 273 (UOF) and 276 ($UF_2$) 10 times each to reveal any ratio variations with sputtering. The summed counts collected at mass 238 (U) varied from 20 000 to 600 000 counts. The mean values of these ion intensity measurements at each of the other masses were evaluated relative to the intensity at mass 238 (U).

The 254 (UO) signals were inherently strong for all samples due to the oxygen primary ion beam that was used. In general, the $^{254}UO/^{238}U$ ion ratio was relatively stable with a variation of a factor 3 between samples. The $^{270}UO_2/^{238}U$ ratio generally tracked the $^{254}UO/^{238}U$ ratio.
In contrast to the uranium oxide ions, the fluorine-containing uranium ions did show significant differences between the type of sample, the storage time and the exposure to ultraviolet light. For samples that were stored in the lab for a time varying between 11 and 16 months, the $^{257}UF/^{238}U$ ratio decreased by a factor 10 compared with the samples that were only 2 months old. Similar decreased values were observed for the $^{273}UOF/^{238}U$ and $^{276}UF_2/^{238}U$ ratios, which generally tracked the $^{257}UF/^{238}U$ ratio. As expected, the $UF_4$ particle sample which was stored in the lab in an unsealed sample box for 29 months did not show any signs of ageing.
To assess the reduction in the level of fluorine by exposure to ultraviolet light, a carbon planchet with particles was exposed to ultraviolet light for 3 weeks. Although the sample was only 2 months old at the time of measurement, the $^{257}UF/^{238}U$ ratio dropped to the same level as the particles that were stored for almost 1 year. These results show that the exposure of $UO_2F_2$ particles to ultraviolet light drastically accelerates the particle ageing process and its accompanied loss of fluorine.

## 4. Discussion

The SIMS measurements described above showed distinct differences between samples with varying storage time and/or UV exposure. However, large particle-to-particle variations within the same sample were observed. These large variations can be attributed to either variations in the particle composition and/or morphology or to SIMS measurement effects related to the ionisation and detection efficiency. In some cases a uniform film of uranium was detected instead of single particles. The fact that the $UO_2F_2$ particles are highly hygroscopic could explain this phenomenon.

## 5. Conclusion and Outlook

The aerosol deposition chamber produces micrometer-sized uranium oxyfluoride particles from the controlled hydrolysis of $UF_6$. These were used for the study of the effect of storage time and ultraviolet light exposure on the relative amount of fluorine in single particles. A link between the relative amount of fluorine and the age of a particle could be established, although large particle to particle variations within the same sample did occur. Transmission electron microscopy (TEM) and Raman measurements will give us more insights into the particle ageing processes.
The observation that the relative amount of fluorine in single particles is an indicator of the time since deposition and possibly the environmental influences could be important in the characterisation of particles collected during safeguards inspections.

## 6. Acknowledgements

The authors acknowledge A. Pidduck and M. Houlton of QinetiQ Malvern, UK for the SIMS measurements and the valuable discussions. Special thanks to S. Van den Berghe and A. Leenaers (SCK•CEN) for the SEM-EDX measurements. The authors would also like to thank O. Marie and F. Pointurier of the Commissariat à l'Energie Atomique (CEA) Bruyères-le-Châtel, France for their support. Many thanks to P. Van Espen, M. Moens and W. Dorriné for the SIMS and SEM measurements at Antwerp University.

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# RECENT ADVANCES IN NUCLEAR FORENSIC SCIENCE

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***Abstract:***

Nuclear forensics has been established as a new branch in science with the continued reporting of seizures of nuclear material and the need for identifying its origin. By today, the IAEA recorded more than 1000 illicit incidents involving nuclear or other radioactive materials [1]. Once illicitly trafficked nuclear material has been intercepted, the questions of its intended use and origin are to be addressed. Especially the origin is of prime importance in order to close the gaps and improve the physical protection at the sites where the theft or diversion occurred. To answer the questions, a dedicated nuclear forensics methodology has been developed. In this paper, the latest challenges will be described. Selected examples shall illustrate the complexity associated with this work. In particular, the need for reference data, for comparison samples will be highlighted, moreover the importance of definition of crime scene procedures and agreed protocols will be underlined. Finally, it has to be noted, that forensics awareness and dedicated training of all actors involved are essential for successful forensic analysis and for possible attribution of the nuclear material.

**Keywords:** Nuclear Forensics, illicit trafficking, nuclear smuggling, chemical impurities, microstructure

## 1. Introduction

Nuclear forensic investigations start after material has been seized and categorized as ‘nuclear material’. They are carried out in order to answer specific questions on the nature of the material and its origin, such as the intended use, the mode of production, the plant and production batch, the last legal owner and the smuggling route. The investigations may comprise conventional forensic tests applied to radioactive material, the morphology of the material, the structure of the material components, the composition of traces in the material and its packing, the isotopic composition of the nuclear material itself and of minor constituents.

Nuclear forensic investigations basically draw upon the information inherent to the material. Nuclear material is generally of anthropogenic origin, i.e. the result of a production process. The nature of this production process is reflected in the elemental and isotopic composition of the material as well as in its microscopic and macroscopic appearance. All of these parameters can be measured using the appropriate analytical technique. Some parameters can be combined to a “nuclear fingerprint”, i.e. they are characteristic for the mode of production of the material. Hence, they may provide a clue on the origin of the material. Nuclear material is either produced by uranium mining which is normally followed by isotope enrichment of uranium or by neutron capture (e.g. in a reactor) which transforms uranium into the transuranium elements neptunium, plutonium etc. The isotopic composition of the latter depends on the reactor conditions and thus allows drawing conclusions on the reactor type and the fuel initially used.

Nuclear forensic investigations have to be considered as part of a comprehensive set of measures for detection, interception, categorization and characterization of illicitly trafficked nuclear material, i.e. as part of a response plan. The Institute for Transuranium Elements (ITU) has been involved in nuclear forensics research since the beginning. During the last ten years, e.g., methods for the age

determination of U and Pu, the geolocation of natural U, and the determination of the reactor type from the Pu isotopic composition have been developed [2-9]. In addition to that, some 30 samples originating from real seizures of nuclear material have been analyzed [10].

## 2. Response Plan and Response Procedures

The detection of nuclear material and the confirmation of the respective alarm generally mark the starting point of the entire response process. This process involves different authorities, each of them working towards their own objectives. A co-ordinated response, however, is essential for successfully handling the case. It is thus essential from the beginning, that all actors involved are aware of the mutual needs and priorities. A response plan needs to be established, defining the roles and responsibilities of the different authorities and actors involved. In a next step, detailed response procedures need to be worked out, describing the manipulations to be performed in order to assure radiological safety and preservation of evidence throughout the entire process, i.e. from the verification of the alarm to the closure of the case.

With regard to nuclear forensics, this means that the preservation of evidence and maintenance of a chain of custody need to be assured at all times. As in classical forensics, also nuclear forensics relies on the fact that certain measurable parameters in a sample are characteristic for the given material. Using these characteristic parameters, one can draw conclusions on the intended use and on the possible origin of the material. This is, however, based on the assumption, that all observed signatures do indeed originate from the material itself. By the same token, this calls for adequate sampling procedures. As in classical forensic investigations, also in nuclear forensics, the experts should, whenever possible, be involved in the investigations from the very beginning.

## 3. Forensic Science and Nuclear Forensic Science

Forensic science (often shortened to forensics) is the application of a broad spectrum of sciences to answer questions of interest to the legal system. This may be in relation to a crime or to a civil action. The use of the term "forensics" in place of "forensic science" could be considered incorrect; the term "forensic" is effectively a synonym for "legal" or "related to courts" (from Latin, it means "before the forum"). However, it is now so closely associated with the scientific field that many dictionaries include the meaning that equates the word "forensics" with "forensic science."
“Forensic” comes from the Latin word “forensis” meaning forum. During the time of the Romans, a criminal charge meant presenting the case before a group of public individuals. Both the person accused of the crime and the accuser would give speeches based on their side of the story. The individual with the best argumentation and delivery would determine the outcome of the case. In other words, the person with the best forensic skills would win [11].

Classical forensics relies on fingerprints, DNA (“genetic fingerprint”), fiber, hair, pollen, residues of explosives or gun-shots. The sampling protocols and the treatment of the samples are well established. Nuclear forensics make use of other parameters, such as isotopic composition of the uranium or plutonium, chemical impurities, macroscopic appearance (e.g. pellet geometry), microscopic parameters (e.g. particle size distribution, grain size distribution, pore size distribution) or the isotopic composition of minor constituents (e.g. Lead or Oxygen). The methodology of nuclear forensics has recently been reviewed [14].

The application of classical forensics to contaminated items is an area that has been addressed only recently. Taking fingerprints or DNA samples from radioactively contaminated pieces of evidence requires an appropriate laboratory environment, that takes radiological aspects into account (protect the analyst from the radioactive material) but allows at the same time to investigate the evidence. A dedicated glove-box was constructed at ITU in collaboration with the German Federal Criminal Police (BKA). The glove-box contains a fuming chamber, for visualizing latent fingerprints using the cyanacrylate method.

In parallel we conducted experiments for determining the radiation stability of the “genetic fingerprint” preliminary results suggest that the DNA can accept fairly high radiation doses before the fingerprint is corrupted. This has been confirmed by other investigators [15].

**Figure 1** Dedicated glove-box for taking fingerprints from contaminated items.

## 4. Examples

Over the past 15 years a wide variety of samples were analysed at ITU in the context of nuclear forensic investigations. Apart from some 30 seized materials, some one hundred samples were analysed for comparison purposes or for research purposes. The following two examples shall illustrate the spectrum of sample types and the challenges associated with each of them.

### 4.1. Uranium Fuel Pellets

In autumn 2006 a series of uranium fuel pellets were received the Institute of Isotopes of the Academy of Sciences (KFKI) in Budapest. The pellets had been seized in Hungary at three different occasions,< three pellets from each of these interceptions were shipped to ITU and were subjected to nuclear forensic investigations in our laboratories in the context of a joint analysis agreement. Consequently, the investigations were carried out in collaboration with two experts from KFKI [12]. In the following example, we focus on one type of pellet, illustrating the attribution process.


**Figure 2** Pellet seized in Hungary and analysed at ITU. This pellet shows markings at the bottom and a slight dishing at the opposite side of the pellet.

The well established sequence of actions was followed, hence first the material was subject to visual inspection, and then the dimensions (diameter and height) were taken. It was noted that the pellets carry a marking (imprint) consisting of five round spots at the bottom. The two opposite sides of the cylindrical pellet show a slight dishing. The pellet shows significant damage, with a lot of material being broken off, particularly at the edges. This is might indicate inappropriate storage and transport conditions, it might also indicate that the pellets were rejected during the production process and considered as scrap (intended to be recycled).

The element content of 88.1% (mass fraction) identifies the material as being $UO_2$. The enrichment, i.e. the $n(^{235}U)/n(U)$ ratio, was measured by high resolution gamma spectrometry. After that, the pellet was dissolved and the isotopic composition of uranium was measured by thermal ionisation mass spectrometry (TIMS) and by multi-collector inductively coupled plasma mass spectrometry (MC-ICP-MS). The results are summarized in Table 1. The pellet consists of natural uranium. The absence of traces of $^{236}U$ indicates that the production facility has not been contaminated with reprocessed uranium, i.e. reprocessed uranium has never been used as feed material in that facility.

| w-% | U-234 | U-235 | U-236 | U-238 |
|---|---|---|---|---|
| HRGS | 0,005 | 0,697 | - | 99,298 |
| TIMS | 0,0049 | 0,711 | - | 99,284 |
| MC-ICP-MS | 0,0051 | 0,712 | <0,0001 | 99,283 |

**Table 1** Isotopic composition of the seized pellet measured by three different techniques

The age of the material is important in order to know the date when the material was produced and thus limit the production period. “Age” is to be understood as the time elapsed since the last chemical purification of the uranium. Such a chemical purification usually precedes the fuel production. The age can be determined through the radioactive decay of uranium to daughter products, hence from so-called parent/daughter ratios. The age of the uranium was calculated using the equation of radioactive decay and its derivatives:

$$N = N_0 \times e^{-\lambda t} \qquad \text{(Eq. 1)}$$

$$\frac{N_{U-234}}{N_{Th-230}} = \frac{N_{0,U-234} \times e^{-\lambda_{U-234} \times t}}{N_{0,U-234} - N_{0,U-234} \times e^{-\lambda_{Th-230} \times t}} \qquad \text{(Eq. 2)}$$

$$t = -\frac{\ln \frac{1-R}{K}}{B} \qquad \text{(Eq. 3)}$$

where

R = measured $^{230}Th/^{234}U$ atom ratio
K = activity ratio $\lambda_{U\text{-}234}/(\lambda_{Th\text{-}230}\text{-}\lambda_{U\text{-}234})$
B = a factor combining the $^{234}U$ and $^{230}Th$ decay constants $(\lambda_{Th\text{-}230}\text{-}\lambda_{U\text{-}234})$

The sample solution was spiked with $^{228}Th$ before U/Th separation [ref.]. The $^{234}U$ concentration was calculated from the previous IDMS results and the $^{230}Th$ concentration was determined from the spiked Th fraction by alpha spectrometry. The age of the material was 16,8 $\pm$ 0,3 a (at the measurement date November 2006), thus the pellets were produced in the end of 1989.

As outlined before [13] ITU operates a database containing information on nuclear fuels (essentially for power reactors), which consists of data from several nuclear fuel manufacturers (including Western Europe and Russia). The data include e.g. dimensions of pellets, $^{235}U$ enrichment and typical impurities. Besides the commercial power reactor fuels the database contains information also on few research reactor fuels and information acquired from open literature. Additionally we insert our research results and results of old findings in the database for a comparison with future cases. Using the information obtained through the measurements mentioned above, a query was put to the database in order to identify the possible origin of the material.

| | | Measured | CANDU | BN-350, BN-600 breeding zone |
|---|---|---|---|---|
| Outer diameter | mm | 12.36 – 0.30 | 12.16 | 13.0 – 0.3 |
| Inner diameter | mm | - | 0.00 | 0.00 |
| Dish diameter | mm | n/a | exists but value unknown | no dish |
| Content $^{235}U/^{tot}U$ | w-% | 0.71 | ≥ natU | < 0.7 |
| Manufacturer | | | various | MZ Elektrostal |
| | | | Possible match | Excluded |

**Table 2** Results of the database query indicating that the pellets were intended for a CANDU reactor.

The results of the query allow to exclude (based on pellet diameter and uranium enrichment) a wide variety of potential applications and production facilities. The only possible match refers to a CANDU type reactor. In Europe there is only one such reactor, located in Cernovada, Romania. The fuel for this reactor was partly imported from Canada and partly produced in Romania, in the Pitesti fabrication plant. By the time the pellets were produced, the Pitesti plant was already operational, hence it cannot be excluded as origin of the material. Availability of or accessibility to reference data or comparison samples from this facility would allow to draw further conclusions.

Pellets of the same characteristics had been seized in Germany at several instances in the years 1992 through 1994.

### 4.2. Uranium powder

While for uranium pellets the geometry and the dimensions provide useful information, uranium powder lacks this attribute and is thus a much more challenging in source attribution. Figure 3 shows three samples of a uranium compound that were investigated. All samples are related to the same incident. For one the three samples only few milligrams of material were made available, while for the two other samples more than a gram was provided.

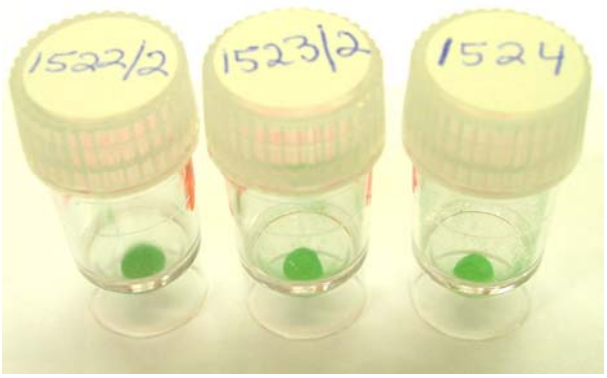


**Figure 3** Three uranium containing samples related to the same incident

The investigation of these samples highlighted a number of challenges. The first challenge was in the verification of the "homogeneity" of the samples, i.e. proving that all three samples originate from the same batch of material or from the same process. To this end, we measured the isotopic composition and the chemical impurities and we investigated the particle morphology.

All samples consisted of natural uranium, the isotope abundance ratios agreed very well within the measurement uncertainty. Also the particle morphology and the particle size distribution agreed very well in the three samples. The chemical impurities, however, allowed interesting observations. Good agreement of the concentration of impurities is found for the "rare" elements, as shown in Figure 4. These data confirm the findings and preliminary conclusion obtained by the isotopic composition and the morphology.

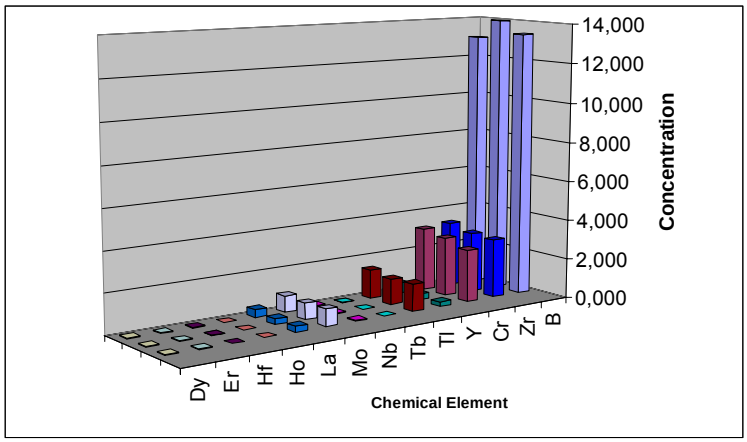


**Figure 4** Concentration of selected chemical impurities [µg/g] in the three samples investigated. Only the "rare" elements are considered in this graph.

Looking at the more common elements, contained as impurity in the samples, we noted a remarkable deviation of one of the samples as compared to the two others. Most of these "omnipresent" elements showed a higher concentration in sample No.1 than samples 2 and 3, as can be seen from Figure 5. At the same time we have to take into account that for sample 1 only few milligrams were provided. An analytical error was excluded by independent repeated analysis (confirming the results) and by blank runs (demonstrating the absence of contamination). This observation suggests that this particular sample was contaminated with common elements during the sampling or sub-sampling process, prior to the arrival in the laboratory. This may have arisen from poorly cleaned sampling tools or sample vials. Obviously, this distortion of parameters could have been avoided by involving a nuclear forensics expert at a much earlier stage of the investigation. It also highlights the importance for the nuclear forensic analyst to obtain as much as possible information on the sample history and the conditions of the seizure.

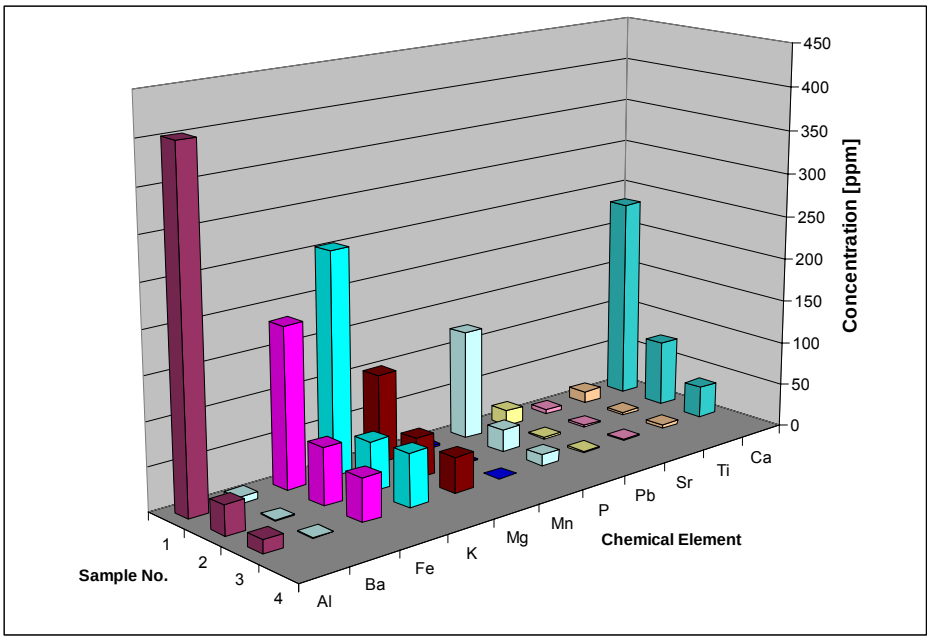


**Figure 5** Concentration of selected chemical impurities [µg/g] in the three samples investigated. Only the "common" elements are considered in this graph.

As the nuclear materials database does not contain information on powders or intermediate products, an attempt for source attribution had to be done using comparison samples. A large series of comparison samples from different (known) origins was investigated and the data were compared against the results of the above samples.

One of the comparison samples matched very well with the material under investigation. All other samples could be rejected on basis of the particle morphology.

## 5. Associated Activities

### 5.1 Training

The issues of illicit trafficking and nuclear forensic science are fairly new. Consequently, the experience gained so far needs to be compiled and analysed. The challenges associated with these issues need to be addressed at different levels: regulatory, scientific, law enforcement. However, the proper responses will only be established if nuclear forensic awareness is developed. Furthermore, the appropriate response procedures can only be applied in the field, if technical training has been provided. Finally, credible nuclear forensics conclusions can only be obtained if the scientists involved apply the best methodology.

As pointed out above, the development of a response plan and the implementation of response procedures are essential for a co-ordinated response and for successfully conducting the nuclear forensic investigations. Dedicated training courses we held in cooperation with the IAEA on this subject with a number of western Balkan countries, with Mediterranean countries and with new EU member states. In the framework of a multi-country TACIS project on combating illicit trafficking of nuclear and other radioactive material similar training courses with Russia, Moldova, Ukraine, Georgia and Azerbaijan are foreseen. While the development and implementation of a response plan consists only the first step, further training for first responders (i.e. customs officers, police and border police) is being provided. Co-ordination of training activities has to be ensured, as such training is offered by several organizations. Increasing the forensics awareness is of paramount importance for preservation of evidence at the crime scene. Therefore, a nuclear forensics awareness training was offered (jointly with the Forschungszentrum Karlsruhe – Fortbildungzentrum für Technik und Umwelt) to law enforcement, measurement institutions and regulatory authorities. Specific technical training in nuclear forensic techniques is also provided in the framework of collaboration under the TACIS programme. Provision of training of measurement (detection) teams is being organised in collaboration with the German authorities.

### 5.2 International Collaboration

Nuclear forensic science is closely related to the phenomenon of illicit trafficking, thus to nuclear security and nuclear safeguards. A border crossing threat is associated with it, hence calling for an internationally co-ordinated response. The International Technical Working Group on combating nuclear smuggling (ITWG) was established some ten years ago, in order to advance the science of nuclear forensics for attributing nuclear material. This is achieved by exchange of information, by developing procedures and recommendations and by exercises.

A number of bi- or multilateral assistance programmes have been set up in order to improve the detection capabilities and to arrange for nuclear forensic assistance. Also the International Atomic Energy Agency (IAEA) promotes the development of nuclear forensics and facilitates the provision of assistance to requesting states which do not have own nuclear forensic capabilities. This is supported by a comprehensive training programme and by a co-ordinated research programme.

The exchange of information on nuclear materials as well as on analytical methodologies is often restricted, due to commercial sensitivities and for national security reasons. Overcoming these restrictions and establishing a broad international cooperation appears highly recommendable in view of the threats of nuclear terrorism, which is unavoidably linked to illicit trafficking of nuclear material.

## 6. Conclusions

In the last 15 years we have seen the emergence of a new and potentially hazardous form of smuggling, namely smuggling of nuclear and radioactive materials. This has triggered the development of a new scientific discipline - nuclear forensic science, where the main aim of the investigations is to find out the origin of the diverted nuclear material; thus allowing to reinforce preventive measures. Existing analytical techniques as used in material science, nuclear safeguards and environmental analysis, were adapted to the specific needs of nuclear forensic investigations. Additionally, new methodologies were developed, aiming at identifying other useful “nuclear fingerprints” in order to reduce the ambiguities often remaining in the interpretation of the analytical results. The lessons which past cases of nuclear forensic investigations taught us, help to constantly improve the forensics procedures, from the “crime scene” to the laboratory. These scientific activities are encompassed by a comprehensive training programme which is implemented by ITU and further developed with competent partners in the field. The hazard involved with nuclear smuggling and the potential relation with nuclear terrorism are the driving forces for deploying and further improving nuclear forensic science and for fostering international collaboration.

## 7. Acknowledgements

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# Session 21

# Plants (reactors)

# Safeguards aspects of recycling plutonium into MOX fuel in light water reactors

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***Abstract***

*Currently, there is an increased interest in nuclear power and several countries are preparing for further nuclear expansion. This will put new demands and standards to safeguards that are necessary for the public acceptance. Additionally, a more efficient use of the natural resources has made the closed fuel cycle more attractive, including the reprocessing of plutonium in spent nuclear fuel for subsequent use as MOX fuel in nuclear reactors. Some of the safeguards issues related to the recycling of plutonium, being a potential nuclear-weapons material, are discussed in this work.*

*Plutonium does not occur in nature but is only produced, in substantial quantities, as a by-product in nuclear power reactors. It is by opponents argued that the reprocessing implies an increased proliferation risk. This work concludes that utilizing the reprocessing cycle increases the number of possible diversion points. However, it is also argued that the composition of the plutonium from spent fuel with a burnup typical for commercial reactors is by no means attractive for a weapons producer. A nuclear device would require the plutonium to be enriched in the fissile isotopes $^{239}$Pu and/or $^{241}$Pu. However, as the quality of the plutonium in spent fuel degrades with higher burnup, fuel with a very low burnup should deserve extra attention. Here, verification tools for burnup and irradiation history are important.*

*MOX fuel can also be assembled of plutonium following the disarmament of existing nuclear weapons, thereby providing an attractive method of reducing the world's inventory of weapons-grade plutonium. All activities and transports of fresh MOX fuel, produced from such plutonium, require, from a safeguards point of view, special attention until irradiation in a reactor effectively has degraded the plutonium to about the same standard as conventional spent fuel. Accordingly, methods to determine burnup of the discharged MOX fuel are important so to verify that the weapons-grade plutonium has been degraded.*

*In this work, experimental techniques are presented for verification of burnup and irradiation history, based on high-resolution gamma-ray spectroscopy. Also, this technique may be used for discriminating between MOX and UOX fuel. In addition, a tomographic technique for integrity verification is presented.*

**Keywords:** Reprocessing, MOX fuel, Plutonium, Low burnup, Safeguards

## 1 Nuclear fuel cycles

Reprocessing plutonium for use in MOX fuel provides efficient use of natural energy resources and thus offers benefits from the perspective of sustainable development. Moreover, the method implies a reduction of nuclear wastes, and accordingly decreases the need of repositories, a fact that has made the United States reconsider their policy about recycling [1].

The reprocessing technique is old and is successively used by several countries, e.g. France and Japan. However, there are still some doubts of the method and scepticism of the benefits. For example, the economics of recycling are questionable even though uranium prices have increased about a factor 10 times over the last few years, thereby levelling out the differences [2]. There are also

concerns about proliferation risks associated with reprocessing, mainly related to possible use of separated plutonium.

The open fuel cycle is comprised of mining, enrichment, fuel fabrication, reactor operation and final storage. From a safeguards point of view, the perhaps most critical step is the enrichment technology for its ability to produce material suitable for nuclear weapons, i.e. Highly Enriched Uranium (HEU). Another key point is the post-reactor procedures since they deal with fuel assemblies containing plutonium.

The closed fuel cycle, schematically illustrated in Figure 1, additionally involves reprocessing of spent fuel assemblies for subsequent MOX-fuel fabrication and reactor operation using MOX fuel. The reprocessing is by opponents regarded as a delicate safeguards issue, as it involves plutonium in a separated form. However, this paper stresses that typical spent fuel has a plutonium content that by no means is optimal for weapons production. The exception to this is fuel with a very low burnup.

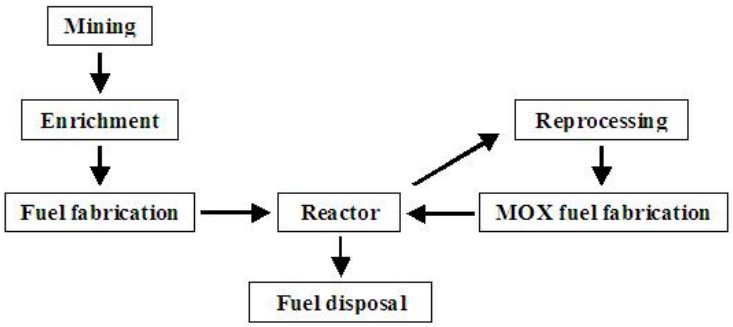


**Fig. 1**: A schematic illustration of the closed fuel cycle. In the steps of reprocessing and fabrication of MOX fuel, plutonium occurs in a separated form.

Moreover, a third fuel cycle can also be recognized as MOX fuel may also be assembled from plutonium with its origin in nuclear weapons. This provides an attractive method of reducing the world's inventory of weapons plutonium while producing energy.

## 2 Plutonium

### 2.1 Plutonium production

Plutonium is not a naturally occurring element, but is only produced in significant quantities in nuclear reactors. The production of $^{239}$Pu requires one single neutron capture event in $^{238}$U (followed by beta decay), and consequently its production starts as soon as the reactor is started. The production of other plutonium isotopes requires repeated neutron capture processes:

- **$^{238}$Pu**: Repeated neutron capture in $^{235}$U. $^{237}$U decays to $^{237}$Np ($\beta^-$, $T_{1/2}$=7 days), which in turn captures a neutron, leading to $^{238}$Np that decays to $^{238}$Pu ($\beta^-$, $T_{1/2}$=2 days).
- **$^{239}$Pu**: Neutron capture in $^{238}$U leads to $^{239}$U that decays to $^{239}$Np ($\beta^-$, $T_{1/2}$=23 minutes), which in turn decays to $^{239}$Pu ($\beta^-$, $T_{1/2}$=56 hours).
- **$^{240}$Pu**: Neutron capture in $^{239}$Pu.
- **$^{241}$Pu**: Neutron capture in $^{240}$Pu.
- **$^{242}$Pu**: Neutron capture in $^{241}$Pu.

Accordingly, in fuel with very low burnup, the plutonium content is almost exclusively $^{239}$Pu. The longer time the irradiation continues the larger share of other isotopes. Similarly, in a higher neutron flux, and hence at a higher power rate, other isotopes are produced more rapidly.

As discussed in section 2.3, the $^{239}$Pu isotope is preferred in nuclear explosives. Therefore, weapons plutonium is produced in specially designated reactors providing low neutron flux, and accordingly low thermal power. Moreover, these reactors are constructed to provide on-line refuelling to optimise the production of $^{239}$Pu. The isotopic composition of a conventional power-producing reactor is degraded

under normal operation conditions, i.e. the content of heavier isotopes such as $^{240}$Pu is significant. The exception to this is fuel with very low burnup.

Theoretically, it is possible to enrich the plutonium in the $^{239}$Pu isotope. However, the enrichment process is based on the weight difference between the isotopes and the differences between the isotopes $^{238}$Pu, $^{239}$Pu and $^{240}$Pu are small. For example, the enrichment of $^{235}$U in natural uranium is performed more easily because the mass difference from $^{238}$U is larger, and yet this is a fairly complicated process. Accordingly, plutonium enrichment of larger quantities is currently not performed.

## 2.2 The properties of the plutonium isotopes

The plutonium isotopes of main interest and some of their properties are summarized in Table 1. It can be noted that fast neutrons can induce fission in all plutonium isotopes. In addition, $^{239}$Pu and $^{241}$Pu are fissile, i.e. fission can be induced by thermal neutrons, and these isotopes can thus be used as fuel in a thermal nuclear reactor as is. $^{238}$Pu, $^{240}$Pu and $^{242}$Pu on the other hand are fertile and can become fissile by neutron capture.

The even-numbered isotopes also undergo, to a large extent, spontaneous fission and thereby they continuously emit neutrons that may induce fission in other nuclei. $^{241}$Pu does not fission spontaneously and therefore emits no neutrons. On the other hand, it decays to $^{241}$Am, which is a strong gamma emitter. Accordingly, samples including $^{241}$Pu will emit more and more gamma quanta with time.

Moreover, $^{238}$Pu and $^{240}$Pu cause substantial heating of the material, even in very small quantities, because they are also strong emitters of alpha particles. The heat is generated as the short-ranged alpha particles are stopped in the material.

| *Isotope* | *Half-life [y]* | *Dominant decay mode* | *Neutron emission [x10$^5$/s kg]* | *Heat [W/kg]* | *Fast fission* | *Thermal fission* |
|---|---|---|---|---|---|---|
| $^{238}$Pu | 87.7 | $\alpha$ | 26.6 | 568 | Yes | No |
| $^{239}$Pu | 24 131 | $\alpha$ | 0.00023 | 1.92 | Yes | Yes |
| $^{240}$Pu | 6 570 | $\alpha$ | 9.1 | 7.1 | Yes | No |
| $^{241}$Pu | 14.4 | $\beta^-$ | - | 3.2 | Yes | Yes |
| $^{242}$Pu | 376 300 | $\alpha$ | 16.9 | 0.113 | Yes | No |

**Table 1**: Properties of some plutonium isotopes.

## 2.3 Usability of plutonium mixtures in nuclear weapons

The mixture of the plutonium plays a fundamental role in the making of nuclear weapons. All mixtures of plutonium could theoretically be used to assemble a nuclear explosive device, since all isotopes of plutonium are fissionable by fast neutrons [3, 4]. However, the conclusions in [3, 4] are drawn from a purely mathematical standpoint, without involving a practical engineering perspective.–From the properties in Table 1, it can be concluded that the most suitable plutonium isotope for a nuclear device is $^{239}$Pu. There are several reasons for why the even-numbered isotopes and $^{241}$Pu are not practicable in nuclear weapons, the most important are:

- Neutron emission from spontaneous fission
- Heating from alpha decay
- Radiation hazard from gamma emission

The main difficulty of the even-numbered isotopes is related to the neutron emission, which may lead to pre-initiation of the fission chain reaction and thus a less optimized and less strong detonation. The strongest neutron emitter is $^{240}$Pu. This has led to a classification [5] of the quality of a plutonium mixture, Table 2, which is based on the relative content of $^{240}$Pu.

| *Grade* | *$^{240}Pu$* |
|---|---|
| Super grade | < 3 % |
| Weapons grade | 3-7 % |
| Fuel grade | 7-18 % |
| Reactor grade | 18-30 % |
| MOX grade | > 30 % |

**Table 2**: Classification of plutonium mixtures for explosive purposes. [5]

Besides the neutron emission, the heat generated from the decay of the isotopes also poses difficulties from an engineering point of view. The decay heat in reactor grade plutonium, depending on composition, is about an order of magnitude greater than in weapons grade plutonium. A sphere of 8 kg reactor grade plutonium would get a surface temperature of 190° C [6], compared to 60° C for a similar sphere of weapons grade plutonium. (In fact, because reactor grade plutonium has a larger critical mass, an even larger difference in temperature can be foreseen.) The high temperatures can lead to difficulties in the handling of the material, but of even more importance is that many of the explosives typically surrounding the plutonium in a nuclear device has a breakdown temperature of about 100° C [3]. Accordingly, especially $^{238}$Pu is non-desirable in this context while $^{239}$Pu offers acceptable levels of the decay heat.

Finally, the use of $^{241}$Pu would with time imply a radiation hazard to the personnel, as it decays to the gamma-emitting $^{241}$Am isotope. On the other hand, this should not affect the usability in a nuclear weapon from a physical point of view.

Whether or not practically possible to assemble a nuclear weapon from various compositions, the most attractive for a possible diverter of weapons material is indeed plutonium with as much as possible of $^{239}$Pu, i.e. super- or weapons grade. It can be noted that all nuclear tests performed up to date have been done with super- or weapons grade plutonium. The exception is a test performed by the USA in 1962 where the exact composition of the plutonium is not exactly known, probably just over 7% $^{240}$Pu [5].

An alternative classification of the plutonium is also proposed in [5] as presented in Table 3. It is based on both mathematical assumptions and on practical engineering limitations, e.g. heat and radiation. Its purpose is to focus the safeguards on plutonium with low timeliness. The high-grade plutonium is conservative and covers both weapons grade as well as much of the fuel grade categories. The low-grade definition basically covers all the reactor-grade plutonium. An analogy to this distribution is the uranium classification with Natural Uranium, LEU (< 20% $^{235}$U) and HEU (> 20% $^{235}$U).

| *Categories* | *$^{240}Pu$* |
|---|---|
| High grade | < 17% |
| Low grade | 17-30% |
| Depleted grade | > 30% |

**Table 3**: An alternative classification of plutonium mixtures for explosive purposes. [5]

## 3. The plutonium content of nuclear fuel from commercial reactors

As pointed out above, several plutonium isotopes are produced during normal reactor operation with different abundances depending on the fuel fabrication and on how the reactor has been operated. This section presents the results of simulations performed with ORIGEN-ARP [7] of how the isotopic composition changes with burnup for different types of fuel.

### 3.1 Isotopic composition of spent LEU fuel

The isotopic composition of the plutonium in light water reactor fuel depends on the power history in the reactor. Calculations of the isotopic composition in the fuel at different burnup have been performed with ORIGEN-ARP [7] for two different values of the initial enrichment, as presented in Table 4. Five cycles of 335 days were simulated, each resulting in a burnup of 10 MWd/kg and accordingly a total burnup of 50 MWd/kg. A decay-period of 30 days was simulated after each cycle.

| *Enrichment* | *Burnup [MWd/kg]* | *$^{238}Pu$* | *$^{239}Pu$* | *$^{240}Pu$* | *$^{241}Pu$* | *$^{242}Pu$* | *Pu-tot.* |
|---|---|---|---|---|---|---|---|
| 3.0% | 10 | 0.2 | 78.7 | 14.3 | 6.2 | 0.6 | 0.43 |
| | 20 | 0.6 | 65.5 | 20.4 | 11.1 | 2.4 | 0.71 |
| | 30 | 1.3 | 56.1 | 23.8 | 13.7 | 5.1 | 0.91 |
| | 40 | 2.2 | 49.8 | 25.6 | 14.5 | 7.9 | 1.05 |
| | 50 | 3.1 | 45.4 | 26.4 | 14.5 | 10.6 | 1.16 |
| 4.0% | *10* | 0.2 | 82.5 | 12.0 | 5.0 | 0.3 | 0.40 |
| | 20 | 0.6 | 70.4 | 17.6 | 9.8 | 1.6 | 0.68 |
| | 30 | 1.2 | 61.1 | 21.3 | 12.9 | 3.5 | 0.89 |
| | 40 | 2.0 | 54.2 | 23.6 | 14.3 | 5.9 | 1.04 |
| | 50 | 3.0 | 48.9 | 25.0 | 14.7 | 8.3 | 1.16 |

**Table 4**: Plutonium isotopic composition in percent at different values of the burnup as calculated with ORIGEN-ARP. The total share of plutonium is presented in percent in the last column.

At a burnup of about 5 MWd/kg, or after half of the first cycle, the share of $^{240}Pu$ increases to more than 7% and hence the plutonium goes from weapons grade to fuel grade according to the classification of Table 2. Consequently, fuel should be discharged from a reactor operated at the conditions considered here within 6 months to obtain weapons grade plutonium. A possible diversion scenario could therefore be to stop the reactor within 6 months, remove some fuel rods and possibly replace them with other rods.

At a burnup of about 20 MWd/kg, or after two cycles, the plutonium turns into the low-grade category of the alternative classification of Table 3. Hence, under the reactor operations considered here, the plutonium is in the high-grade category only during the first two cycles. As this is a fairly low level of burnup, these results show that most spent fuel assemblies from normal reactor operations contain low-grade plutonium.

The results are also illustrated in Figure 2 for the fuel with an initial enrichment in $^{235}U$ of 4%. As the figure shows, the total amount of plutonium increases with irradiation to about 1.2% at a burnup of 50 MWd/kg, which corresponds to about 6 kg of reactor-grade plutonium in a typical PWR assembly. At a burnup of 5 MWd/kg there is about 1 kg plutonium, however of weapons grade.

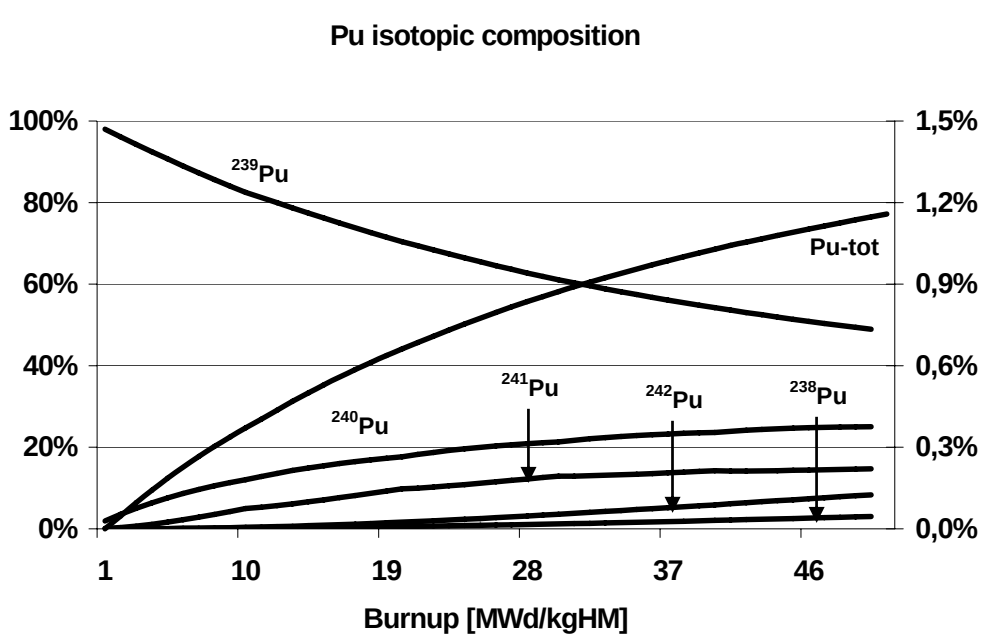


**Fig. 2**: Plutonium isotopic composition at different values of the burnup as calculated with ORIGEN. Also the total amount of plutonium [%] is indicated. The fuel had an initial enrichment in $^{235}U$ of 4%.

### 3.2 Isotopic composition of spent MOX fuel

As MOX fuel is irradiated in a reactor, it will contain even more of the isotopes $^{238}$Pu, $^{240}$Pu and $^{242}$Pu as compared with conventional $UO_2$ fuel (section 3.1). Table 5 and Figure 3 shows the isotopic composition of plutonium in MOX fuel at discharge from the reactor as calculated with ORIGEN-ARP. The fuel was in the simulations initially assembled with 4.3% recycled reactor grade plutonium and 0.25% $^{235}$U. The total amount of fissile material, i.e. $^{235}$U, $^{239}$Pu and $^{241}$Pu, was 3.0%.

| *Burnup [MWd/kg]* | *$^{238}$Pu* | *$^{239}$Pu* | *$^{240}$Pu* | *$^{241}$Pu* | *$^{242}$Pu* | *Pu-tot.* |
|---|---|---|---|---|---|---|
| 0 | 2.5 | 54.7 | 26.1 | 9.5 | 7.2 | 4.30 |
| 50 | 3.0 | 32.9 | 29.6 | 16.8 | 17.7 | 2.85 |

**Table 5**: Plutonium isotopic composition in percent for spent MOX fuel from reprocessed reactor grade plutonium. The total share of plutonium is presented in percent in the last column.

It can be noted that the total plutonium content in the irradiated MOX fuel still is as much as about 2.9% as compared to the initial 4.3%. Accordingly, the total plutonium content is only decreased with about one third. However, the isotopic composition is even more degraded as the fissile material decreases while other isotopes, e.g. $^{240}$Pu, $^{242}$Pu and $^{238}$Pu increases. This makes the plutonium less attractive for further use in reactors or, especially in the safeguards context, nuclear weapons. The plutonium is on the border to be classified as depleted grade in the alternative classification of Table 3.

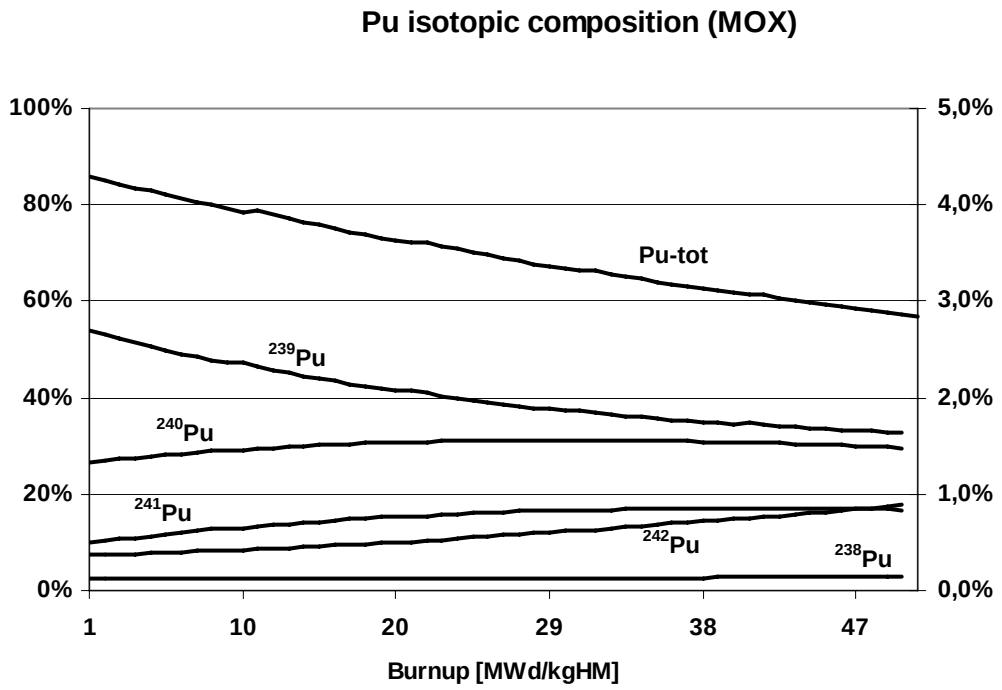


**Fig. 3**: Plutonium isotopic composition of MOX fuel at different values of the burnup as calculated with ORIGEN. The total plutonium content is also illustrated (right axis).

### 3.3 Isotopic composition of spent MOX fuel of WG plutonium

A powerful way of eliminating weapons-grade plutonium, as a part of the decommissioning of nuclear warheads, is to make MOX fuel of it and use it in a reactor for power production. During the irradiation, the relative content of $^{239}$Pu is lowered while $^{240}$Pu and higher isotopes are increased. Moreover, the irradiation leads to a net reduction of approximately 30% in the initial amount of plutonium. Table 6 presents compositions [8] for typical operations with a burnup up to 40 MWd/kg:

| *Fuel* | $^{239}Pu$ | $^{240}Pu$ | $^{241}Pu$ | $^{242}Pu$ |
|---|---|---|---|---|
| MOX initial | 93.5 | 6.5 | | |
| MOX 40 MWd/kg | 51.0 | 29.0 | 16.0 | 4.0 |

**Table 6**: Plutonium isotopic composition of MOX fuel from weapons grade plutonium, given in percent initially and after a burnup of 40 MWd/kg [8].

## 4. Experimental techniques for verification of burnup, irradiation history and fuel integrity

Based on the information in sections 2 and 3, it can be argued that if diversion of plutonium were attempted, low burnup material would be the most attractive target. Hence, extra attention of such material can be justified, and in this context, verification tools for burnup and irradiation history are important. In addition, methods to discriminate LEU fuel from MOX fuel can also be useful. Moreover, it may be relevant to perform integrity verification of fuel assemblies with low burnup, in particular if there is low burnup MOX fuel that has been assembled from weapons-grade plutonium. In this section, some experimental techniques addressing these issues are presented.

### 4.1 Verification of burnup and irradiation history

Some techniques for verifying fuel burnup are described in [9] including measurements of both gamma quanta and neutrons. In gamma-ray measurements, the intensity of $^{137}Cs$ ($T_{1/2}$ = 30 years) is typically used, because its production is linear with burnup. Neutron sources on the other hand have a more complex production, where the most dominant source, $^{244}Cm$, increases with burnup roughly to the power of four. This non-linear dependence complicates the use of neutrons for burnup verification, especially for fuel with low burnup.

A method that simultaneously allows for the verification of also the irradiation history and cooling time, would offer even better safeguards properties. Such a technique is under development at Uppsala University [10] based on high-resolution gamma-ray spectroscopy (HRGS). Here, not only $^{137}Cs$ is analysed but also other long-lived fission products are considered, i.e. $^{134}Cs$ ($T_{1/2}$ = 2 years) and $^{154}Eu$ ($T_{1/2}$ = 8 years). By controlling that the measured ratios of these isotopes are consistent with the declared burnup, irradiation history and cooling time, verification of these data can be performed. Here the half-lives of the isotopes are used together with their respective buildup dependencies. In the measurements performed so far [10], a fuel assembly is scanned axially by moving it vertically in front of a collimator mounted in the pool wall. The gamma detection system is situated on the dry side of the wall. The measurement setup is schematically illustrated in Figure 4. It would also be possible to use a submergible measuring device in the fuel pool [11].

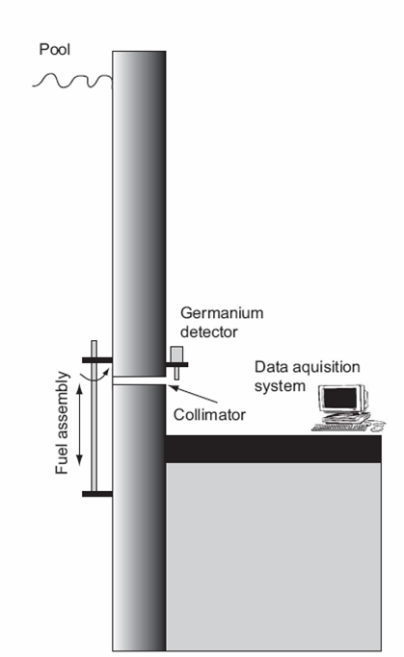


**Fig. 4**: Schematic view from the side of the gamma scanning setup used in [10]. The fuel assembly is scanned by moving it vertically in front of a collimator mounted in the pool wall. The assembly can also be rotated around its axis for measurements at different angles.

Using the equipment of Figure 4, the burnup and cooling time of LEU fuel has been verified within 1.6% and 1.5%, respectively [10] and also the irradiation history was found to be consistent with operator-declared data.

The method should also be applicable to MOX fuel, where one may note that reprocessed plutonium generally has an isotopic composition that refers to relatively high burnup and therefore the demands on accuracy are somewhat relaxed. However, if the MOX fuel is manufactured from weapons-grade plutonium, it is highly relevant to verify its burnup to assure that the plutonium has been degraded.

## 4.2 Discriminating MOX/LEU fuel

The HRGS technique described above may also be used to determine whether a fuel assembly is of MOX or LEU type [12]. Here one uses the fact that $^{239}$Pu and $^{235}$U have different yields of the fission product $^{154}$Eu but almost the same yields of $^{134}$Cs. Accordingly, by measuring the ratio of the gamma-ray intensities of $^{134}$Cs and $^{154}$Eu one can obtain a significant difference between MOX and LEU fuel, respectively.

## 4.3 Verification of fuel integrity

The integrity of fuel assemblies is an important safeguards parameter. It may be of particular interest to verify that all fuel rods are present in fuel with low burnup, because the plutonium content of such fuel is of weapons grade, see section 3. Here, the DCVD [13] may be a useful tool. It is based on detecting the Cherenkov light produced in the water surrounding the fuel rods in an assembly. However, there may be limitations in the amount of diverted material that can be confidently detected.

Other methods proposed for this purpose are based on the tomographic technique SPECT [14, 15]. Using these methods, it is possible to obtain information of the interior of a fuel assembly at any selected axial level by external measurements. An example of a cross-sectional image obtained in such a measurement is presented in Figure 5 [15]. The measurement was performed on a SVEA-96S fuel assembly at the Swedish NPP Forsmark 2. The fuel assembly had relatively low burnup (about 10

MWd/kgU), as it had only been irradiated for one power cycle of about one year at the time of the measurement. As seen in the image, the fuel rods can be clearly distinguished from the background.

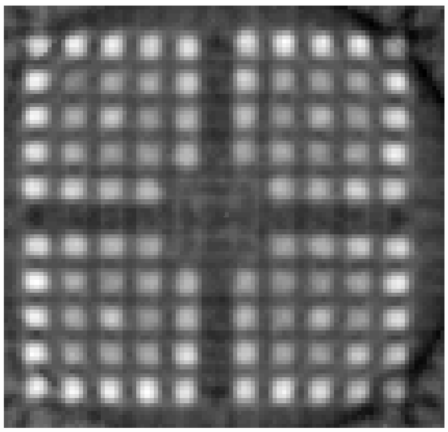

**Fig. 5**: A cross-sectional image obtained in the measurement of a SVEA-96S fuel assembly at the Forsmark 2 NPP in 2002.

In order to evaluate the applicability of the technique for detecting the removal of individual fuel rods, simulations of such fuel have also been performed. An example of an obtained cross-sectional image reconstructed from the simulated data is shown in Figure 6.

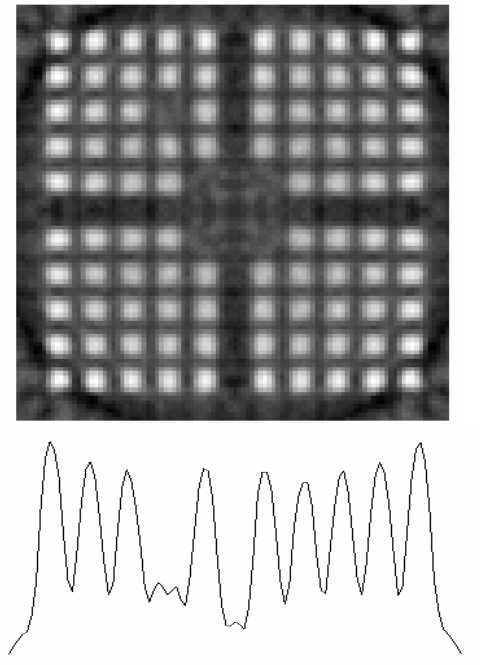

**Fig. 6**: Obtained cross-sectional image (top) of a simulated SVEA-96S fuel assembly with one fuel rod removed. As shown in the graph of activities in the third row (bottom), the missing rod can be clearly distinguished.

The results indicate that the tomographic technique can be very useful for verifying fuel integrity on the individual fuel rod level.

## 5 Conclusions and discussion

Considering the properties of the different plutonium isotopes, the most attractive material for constructing a nuclear device is super grade or weapons grade plutonium, i.e. predominantly $^{239}$Pu

containing less than 7% $^{240}$Pu. According to the simulations of fuel composition presented in section 3, this refers to UOX fuel assemblies with low burnup, typically less than 5 MWd/kg or half a year of irradiation in a commercial power-producing reactor. Using an alternative definition of plutonium mixtures, high-grade plutonium can be defined as plutonium containing less than 17% $^{240}$Pu, corresponding to a burnup of less than 20 MWd/kg or two years of irradiation. It may also be noted that MOX fuel manufactured from reprocessed plutonium typically contains plutonium of lower grade. On the other hand, MOX fuel may also be manufactured from weapons plutonium, thus containing plutonium of higher grade.

In this context, it may be of interest to verify the fuel burnup in order to control that the plutonium composition is not suitable for nuclear weapons production. Also, it may be of interest to control whether a fuel assembly is of the UOX or the MOX type. It has been argued in this paper that both these issues can be addressed using the HRGS technique. In addition, it may be relevant to verify the integrity of fuel assemblies with low burnup, to control that weapons-grade plutonium is not being diverted. For this purpose, the tomographic technique has proved to be a promising verification tool.

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# Field Application of a Portable Detector for the Verification of Research Reactor Spent Fuel

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## Abstract

The advanced experimental fuel counter (AEFC) [1] was developed for the measurement of spent fuel rods and assemblies from research reactors for safeguards verification. This measurement system contains components for active neutron interrogation, passive neutron totals counting, neutron coincidence counting, and gross gamma-ray counting. For measuring the $^{235}U$ fissile mass, the active neutron interrogation component has an AmLi neutron source. The active assay mode uses two measurement methods: 1) neutron coincidence counting, and 2) totals neutron differential transmission, in which the interrogation source (AmLi) has lower average neutron energy than the induced fission neutrons. The AEFC was used for measurements of spent fuel arising from operation of the heavy water cooled and moderated research reactor (HIFAR) in Australia on Nov 10-14, 2006. This was the first application of the AEFC for verification of spent fuel enrichment/fissile content etc, and the measurements were made by participants from the IAEA, ASNO (Australian Safeguards and Non-proliferation Office), ANSTO (Australian Nuclear Science and Technology Organization), and LANL. The paper presents the results of the first application of the AEFC for spent fuel at the research reactor in Australia.

**Key Words:** spent fuel, research reactors, NDA instrumentation, neutron detectors

## 1. Introduction

There are many research reactors worldwide that have been, or are being converted from the use of high enrichment uranium (HEU) to low enrichment uranium (LEU, < 20% enriched). The verification of fissile content and enrichment the spent fuel is needed for the effective safeguards of the fuel by the inspectorate. The AEFC was developed for the measurement of plutonium and U-235 content and enrichment properties of spent fuel rods and assemblies from research reactors and MAGNOX (magnesium oxide clad fuel) reactors. This measurement system contains components for active neutron interrogation, passive neutron totals counting, neutron coincidence counting, and gross gamma-ray counting. For measuring the $^{235}U$ fissile mass, the active neutron interrogation component has an AmLi neutron source, similar to the one used in the Active Well Coincidence Counter (AWCC) [1] and the Research Reactor Fuel Counter [2]. The active assay mode uses two measurement methods: 1) neutron coincidence counting and 2) totals neutron differential transmission, in which the interrogation source (AmLi) has lower average neutron energy than the induced fission neutrons.

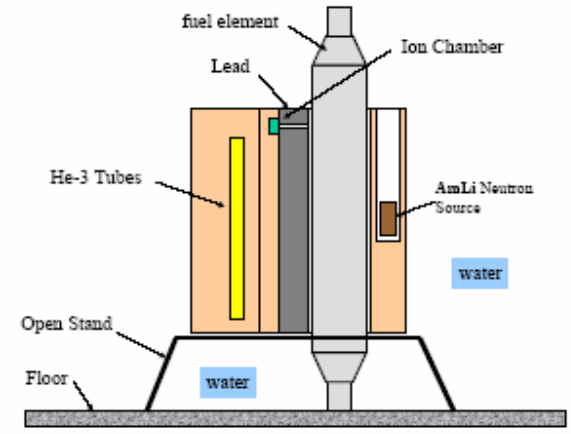
Overall, seven detectors are used. The four $^{3}He$ tubes near the sample position are for coincidence counting, whereas the two $^{3}He$ tubes furthest from the sample are for the differential

transmission totals neutron measurement. The passive neutron coincidence measurement of the sample provides the spontaneous fission neutron rate from the $^{240}$Pu-effective in the samples. The gross gamma-ray activity from the sample is measured with a small collimated ion chamber located near the top of the detector package. The burnup profile of the fuel element is measured as the fuel is entered and removed from the system. This report covers the AEFC initial calibration at LANL [3], and the application to spent fuel of the DIDO type at the HIFAR spent fuel pool in Australia.

## 2. System Description

The AEFC consists of a cylindrical polyethylene moderator, lead shielding, and detector tubes surrounded by a stainless steel body. The sample through-hole diameter is 117 mm. Figures 1 and 2 show the configuration of the internal components and Fig. 3 shows the finished package. The six $^{3}$He tubes have a $^{3}$He fill pressure of 4 atm. The tubes are expected to measure neutrons without gamma interference for sample surface dose levels up to 5000 R/h (50 SHV/h). Higher dose levels can be measured by reducing the HV bias on the He-3 tubes and operating at a reduced efficiency. The ion chamber near the top of the system can provide a measurement of the fission product gamma-rays and a burnup profile as the spent fuel is moved through the sample channel.

The neutron data was collected using a JSR-12 that would normally be connected to a lap top computer. However, the interface cable between the computer and JSR-12 was missing in the shipment so the neutron data was recorded by hand to the log book.



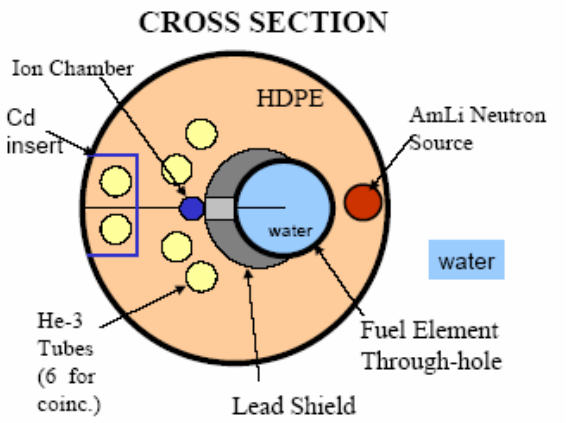




***Figure 1.*** *Side cross-sectional diagram of the AEFC.* ***Figure 2.*** *Top cross-sectional diagram .*

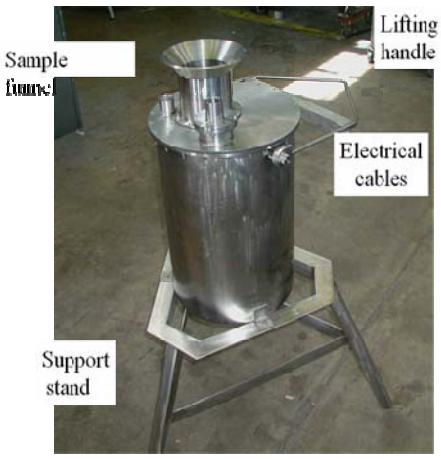



***Figure 3**. Photograph of the assembled AEFC on the support stand.*

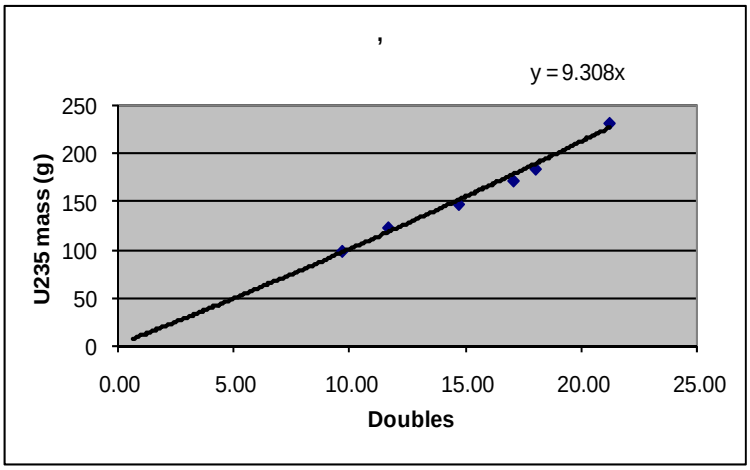



***Figure 4**. AEFC Doubles calibration curve for $^{235}$U in MTR fuel and AmLi source N-074.*

A $^{252}$Cf, source was used to measure the high voltage (HV) plateau, efficiency, and die-away time at LANL. The normal operating HV was 1640V and the efficiency was 4.6%. Since the neutron intensity from research reactor fuel elements is expected to be relatively low, the dead-time correction should be small, and no dead-time corrections were made to the HIFAR data.

## 3. Calibration Approach

The calibration of the AEFC for the DIDO type fuel is complicated by the facts that 1) there were no DIDO fuel element standards available, 2) the AEFC detector length covers only about 50% of the total fuel length (~ 60 cm), and 3) the plutonium and $^{235}$U distributions along the fuel element length are non-uniform due to the reactor flux profile. To address these issues, we have used a fresh MTR fuel element containing a known amount of $^{235}$U to obtain the active and passive neutron rates. The same element has been simulated with MCNPX calculations [4] and the data is used to benchmark the calculations. The MCNPX calculations can then be used to correct the data for the non-uniform vertical fuel distribution.

There are several measurement objectives and multiple measured parameters to provide answers to the inspection questions. The verification objectives for the spent fuel include:

- the residual $^{235}$U mass,
- the burnup, and
- the initial enrichment (LEU versus HEU).

The measured quantities that are obtained from the AEFC are listed in Table 1. The fuel elements are measured in both the passive mode and the active mode with neutron and gamma-ray detectors. After about a year cooling time of the spent fuel elements, most of the short-lived fission products have decayed away and Cs-137 is the dominant gamma source. The ion chamber near the top of the AEFC provides the relative gamma activity as well as the burnup profile. Unfortunately, the gamma-ray leakage on the outside of the lead shielding distorted the gamma profile. Future AEFC units will include additional shielding to correct this problem.

| **Table 1. Measurement signals and spent fuel attributes.** | |
|---|---|
| **Measurements** | **Primary Fuel Attribute*** |
| **Passive Mode** | |
| Singles rate (front 4-tubes) | $^{240}$Pu-e (SF) + alpha,n |
| Doubles rate (front 4-tubes) | $^{240}$Pu-e (SF) |
| Singles rate (back 2-tubes) | $^{240}$Pu-e (SF) + alpha,n |
| Ion Chamber | Relative gamma (Cs137) + profile |
| Ratio Front detectors/back detectors | Average neutron energy |
| **Active Neutron Interrogation Mode** | |
| Net Singles rate (front 4-tubes) | $^{235}$U (IF) |
| Net Doubles rate (front 4-tubes | $^{235}$U (IF) |
| Net Singles rate (back 2-tubes) | $^{235}$U (IF) |
| AmLi neutron source background | AmLi source background |

*The $^{240}$Pu-effective spontaneous fission is abbreviated $^{240}$Pu-e (SF), and induced fission in uranium as $^{235}$U (IF).

The passive mode neutron measurement is the primary signal used to determine the initial enrichment as LEU versus HEU. However, for the HIFAR fuel elements, the (alpha,n) yield changes for the LEU and HEU, because the LEU is in a deposit of uranium silicide, and the HEU is in a UAlx alloy. For high burnup LEU fuel, the curium isotopes might add to the neutron emissions. The passive neutron rate is also a function of the burnup, so the $^{235}$U active neutron measurements are needed to determine the fuel element burnup. The AEFC has redundancy in the $^{235}$U measurement with the front tube Doubles and Singles rates and the back tubes Singles rate.

There is significant neutron multiplication in the fissile content in the fuel element and the MCNPX simulations are used to estimate the multiplication perturbation to the Singles and Doubles count rates. In most cases, the calibration measurement includes the multiplication effect. The active mode calibration curves measured with the MTR fuel elements include the neutron multiplication for the Singles and Doubles rates. The plutonium does not significantly contribute to the multiplication because of its low content per element compared with the $^{235}$U mass per element.

In general, the AEFC measures the central half of the fuel element, and the active mode calibration units are g$^{235}$U /cm. Similarly the passive measurements are in units of gPu-240e/cm. The length of the fuel assembly is used to get the total mass in the element. The fuel length can be confirmed with the gamma scan.

## 4. MTR Fuel Element Active Calibration Measurements

The un-irradiated MTR fuel element L-108 at LANL was measured in the AEFC to obtain the passive and active neutron rates. The element contains 231.7 g $^{235}$U in a parallel plate configuration with an active fuel length of 60 cm. For the calibrations, the measurement set up had the MTR element centered in the active zone of the He-3 tubes for the AEFC in a water tank. The element was rotated by 90 degrees and 180 degrees and the counting rates were the same within the counting precision of ~1-2%. Because the thermal-neutron interrogation has both

self-shielding and multiplication effects in the fuel plates, the calibration function is not a linear function with zero intercept. The shape of the curve comes from the MTR data. The self-shielding and multiplication partially cancel each other in the Doubles curve. Figure 4 shows the Doubles calibration curve for the $^{235}U$ content per unit length. The calibration standard had a uniform fissile density along the length.

Because of the heavy water moderator and reflector for the HIFAR reactor, the burnup profile along the length of the spent fuel element is relatively uniform with a small decrease near both ends. Thus, the ends of the assembly have an increase in the residual $^{235}U$ and a decrease in the $^{240}Pu$-e buildup. The measured active neutron rates were increased by a factor of ~1.05 to correspond to a uniform distribution of $^{235}U$ in the calibration standard.

## 5. HIFAR Spent Fuel Elements

The spent fuel elements at the HIFAR reactor are of the DIDO design with an active length of 60 cm and an outside shell diameter of ~10cm. The cylindrical fuel design has 4 concentric fuel tubes clad in aluminum with an outer annulus of aluminum. The fuel elements consist of several different initial uranium enrichments and alloys. The different fuel types and approximate reactor dates are listed in Table 2 [5,6].

**Table 2. Fuel element loading history for the HIFAR reactor**

| Date | U-235 Enrichment | Composition |
|---|---|---|
| 1962 | 93% | UAlx alloy |
| 1963 | 80% | UAlx alloy |
| 1984 | 60% | UAlx alloy |
| 2004 | 19.75% | U silicide |

The active neutron measurements will determine the residual $^{235}U$ content in the fuel elements without significant dependence on the cooling time and fuel plate composition. However, the passive neutron and gamma measurements will have a dependence on the cooling time and the composition of the fuel tubes. For the active mode assay, the net Singles rates the Doubles rates will not depend on the (alpha.n) yields after the multiplication correction.

The declared specifications for the HIFAR fuel elements that were measured are listed in Table 3.

**Table 3. HIFAR Fuel Element Specifications**

| fuel id | | 18A-382 | ued727 | ed564 | d&e14 | 423-492 | 423-527 | 423-543 |
|---|---|---|---|---|---|---|---|---|
| before irradiation nominal | total U (g) | 911 | 188 | 188.1 | 141.5 | 278 | 280.6 | 287 |
| | $^{235}$U (g) (enrichment) | 180 (19.8%) | 150.4 (80%) | 150.4 (80%) | 113.2 (80%) | 167.9 (80%) | 168.9 (60%) | 172.2 (60%) |
| post irradiation fuel composition | total U (g) | | 151 | 137.6 | 112.5 | 189.2 | 204 | 189.4 |
| | $^{234}$U (g) | | 0 | 0 | 0 | 1.9 | 2 | 2 |
| | $^{235}$U (g) | 88.42 | 106.5 | 90.9 | 79.3 | 67.7 | 82.4 | 62.2 |
| | $^{236}$U (g) | | 6.8 | 9.2 | 5.2 | 32.7 | 31.1 | 34.8 |
| | $^{238}$U (g) | | 37.6 | 37.5 | 28 | 86.9 | 88.5 | 90.4 |
| | total Pu (g) | 5.89 | 0.4 | 0.5 | 0.26 | 1.58 | 1.43 | 1.68 |
| | $^{238}$Pu (g) | | 0.002 | 0.005 | 0.001 | 0.12 | 0.083 | 0.155 |
| | $^{239}$Pu (g) | | 0.36 | 0.42 | 0.24 | 1.06 | 1.04 | 1.06 |
| | $^{240}$Pu (g) | | 0.04 | 0.067 | 0.025 | 0.283 | 0.228 | 0.323 |
| | $^{241}$Pu (g) | | 0.001 | 0.002 | 0 | 0.088 | 0.067 | 0.113 |
| | $^{242}$Pu (g) | | 0 | 0.002 | 0 | 0.021 | 0.011 | 0.031 |
| time in reactor (days) | | | 83 | 84 | 58 | 210 | 210 | 245 |
| cooling time (days at 30/11/06) | | 483 | 14849 | 15521 | 16331 | 1435 | 1120 | 910 |
| burn up (%) | | 44.6 | 29 | 40 | 30 | 60 | 51 | 64 |

## 6. Measurement Method

Each fuel element was first measured in the passive mode, and then the AmLi neutron source was inserted into the AEFC to obtain the active neutron interrogation measurement. The net active neutron measurement is determined by subtracting the passive neutron rates from the element and the AmLi source. Figure 5 shows the AEFC positioned at the bottom of the ~ 5 meter deep` HIFAR pool with a spent fuel element inside the measurement channel. The extra Al cylinders on the floor of the pool are the shells from other fuel assemblies that have had the outer Al shell removed. For the measurements, the fuel elements were rapidly scanned through the AEFC until the element touched the floor. After a short pause (~ 30s) the element was raised by ~ 50 cm from the floor so that the bottom of the fuel was about 1 cm below the bottom of the AEFC. This “reference” position was chosen to provide maximum lead shielding for the He-3 tubes.

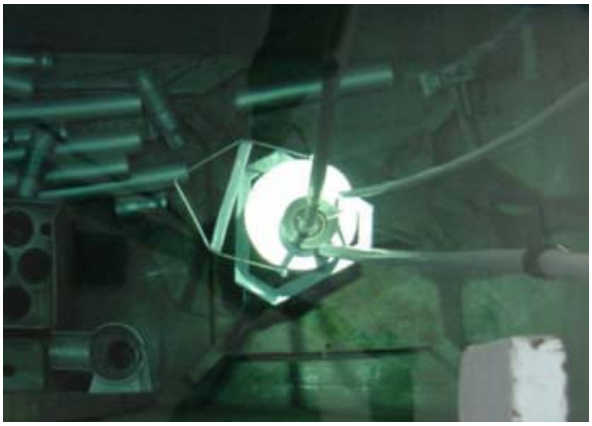


***Figure 5****. Photo of the AEFC on the bottom of the HIFAR spent fuel pool. The flexible tube to the surface contains the electronics cables and the ridged PVC tube contains the AmLi neutron source transfer cable.*

Figure 6 shows a typical scan of the fuel measurement where the top graph is the data from the ion chamber and the lower graphs are the Singles neutron rate from the front tubes and back tubes, respectively. The split signal from the neutron channel was collected continuously in 1 s time steps using the IAEA MiniGRAND. The standard IAEA MIC software was used to collect the data and the RAD software was used to display and analyze the data. The neutron peaks are offset from the gamma peak by about 24 cm. The gamma peak is broader than the neutron peak because of gamma leakage into the ion chamber from above and below the ~ 15 mm diameter collimator hole in the lead.

After the fuel element is in the reference position, the data was collected for about ten minutes to obtain better statistics for the neutron Doubles rates. The flat regions in Fig. 6 at the end of the scans correspond to the stationary fuel in the "reference position".

### 6.1 Active Mode Measurement

For the $^{235}U$ measurement, the fuel element was positioned in the reference location, and the AmLi neutron source was inserted through the PVC tube into the AEFC. The source removal distance to get a negligible background from the AmLi source was ~ 0.7 m.

Figure 6 shows the data for both the passive scan and the active assay period after the AmLi source is inserted into the AEFC about halfway through the displayed data. The increase in the neutron rate is about half from the induced fissions and half from the AmLi neutron source for the front tubes, but the back tubes have a signal to source background of about 10/1, so about 90% of the rate increase comes from the induced fissions in the $^{235}U$. Both of these rates were used to calculate the residual $^{235}U$ mass in the element.

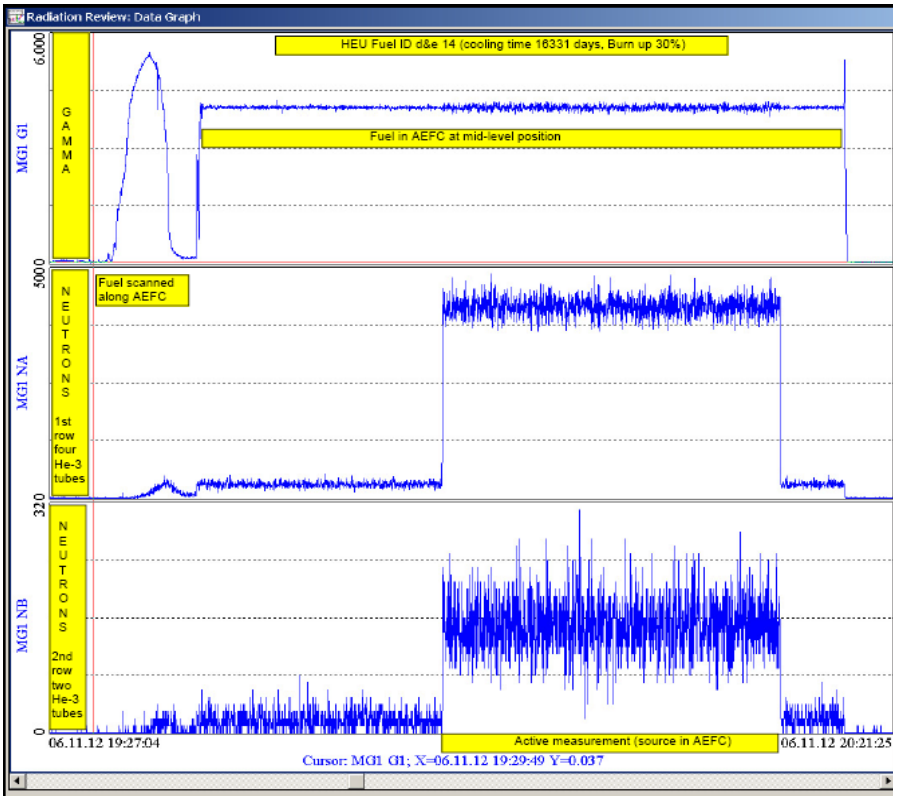



***Figure 6****. Passive and active scan of HIFAR fuel element through the AEFC with a pause at the floor and then raised to the reference position. The top curve is the gammas and the bottom two curves are for the front and back tube neutrons, respectively.*

## 7. RESULTS – ACTIVE NEUTRON FISSILE MEASUREMENTS

### 7.1 Active Mode Spent Fuel Measurement Results For Front Detectors

During the measurement activity, the data was collected in two systems where the neutron coincidence data collected with the JSR-12 and the ion chamber and the Singles neutron data collected with the MiniGRAND (MIC software and RAD review). The signals were connected with a split of the neutron data to the MiniGRAND operating continuously and the JSR-12 was started for each measurement manually because the computer connection was missing.

The neutron coincidence rates from the JSR-12 , after HV corrections, are shown in Fig. 7 versus the declared $^{235}$U. Table 4 lists the active and passive data collected in the MiniGRAND. The back 2-tube data was available from the MiniGRAND scan. For the active neutron measurements, the AmLi background rates of 187 and 1.9 were subtracted from the front and back tubes, respectively. The designation A-4 tubes corresponds to active mode rate in the front 4 he-3 tubes, etc.

Figure 8 shows the net Singles from the back tubes versus the tag $^{235}$U mass. Both the Doubles and singles curves show a good correlation between the measured rates and the declared $^{235}$U

masses. The Doubles curve shows the increase in the multiplication for the higher $^{235}U$ content, however, there is more statistical scatter for the Doubles than the Singles.

Four of the seven HIFAR fuel elements had short cooling times and high gamma dose rates so that the gamma pileup interfered with the neutron counts at 1640V. For these four cases, the data was collected at a high voltage of 1500V. For element ued-727, the gamma dose was low and the data was collected at both 1640V and 1500V to provide a normalization between the two HV settings.

**Table 4. MiniGRAND Measurement Results – Active and Passive Singles Rates**

| Sample | g U235 | A-4 Tubes | P-4 Tubes | Net-4 | A-2 Tubes | P-2 Tubes | Net 2 | Net4 - 187 | Net2 – 1.9 |
|---|---|---|---|---|---|---|---|---|---|
| 382 | 88.4 | 913.69 | 550.06 | 363.63 | 39.64 | 26.04 | 13.60 | 176.6 | 11.7 |
| ued727 | 106.5 | 514.02 | 85.42 | 428.60 | 21.13 | 4.80 | 16.33 | 236.0 | 14.4 |
| D&e14 | 79.3 | 416.34 | 29.61 | 386.73 | 15.00 | 1.66 | 13.34 | 193.7 | 11.4 |
| ed564 | 90.9 | 514.27 | 107.29 | 406.99 | 21.28 | 6.10 | 15.18 | 214.0 | 13.4 |
| 543 | 62.2 | 2320.93 | 1984.26 | 336.67 | 110.70 | 99.95 | 10.75 | 143.7 | 8.85 |
| 492 | 67.7 | 2248.91 | 1887.93 | 360.97 | 104.38 | 94.20 | 10.18 | 168.0 | 8.28 |
| 527 | 82.4 | 1541.06 | 1165.35 | 375.71 | 79.98 | 66.59 | 13.40 | 182.7 | 11.5 |

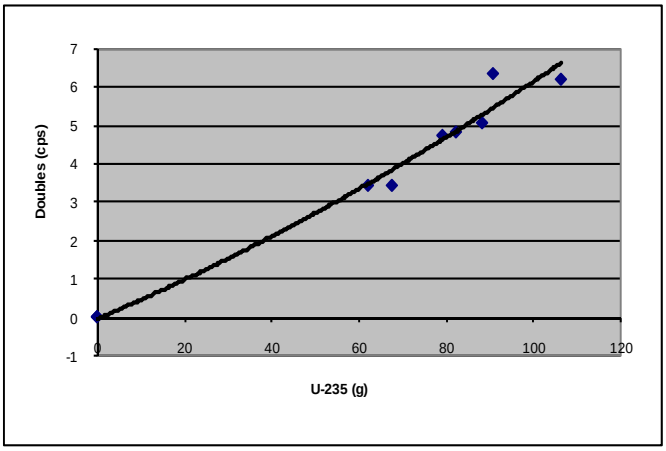



***Figure. 7.*** *Declared U-235 mass versus Doubles in the front He-3 detectors. .*

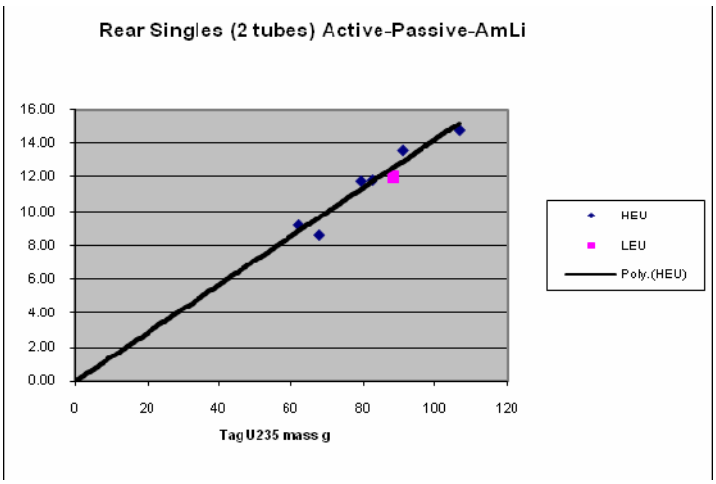



***Figure. 8.*** *Declared U-235 mass versus net Singles in the back He-3 detectors.*

## 7.2 Active Mode Spent Fuel Measurement Results Back Detectors

The alternative measurement of the fissile content comes from the Singles rate in the two back He-3 tubes. The back detectors have a much better signal/background ratio then that of the front detectors and provide diagnostic information related to the neutron energy. Figure 8 shows a

plot of the measured net Singles rate in the back detectors as a function of the declared $^{235}U$ content. The relationship is almost linear.

When we convert the measured Singles rates to $^{235}U$ g using the MTR calibration, we get the $^{235}U$ mass in the fuel assemblies. Table 5 gives the measured $^{235}U$ mass values for the three calibration curves. The RSD between the measured average $^{235}U$ mass and the declared values was 5%.

**Table 5. Active mode measurement results from the MiniGRAND for the Singles and the JSR-12 for the Doubles.**

| Assembly (enrichment) | Active D (4 tubes) cps | Active S (4 tubes) cps | Active S (2 tubes) cps | D-4 U235 g | S-4 U235 g | S-2 U235 g | Ave. U235 g | Decl. U235 g | Ave. Diff. % |
|---|---|---|---|---|---|---|---|---|---|
| 382(19.9%) | 8.58 | 191.9 | 12.71 | 88.8 | 77.3 | 75.7 | 80.6 | 88.4 | -9.7 |
| ued727(80%) | 10.51 | 256.5 | 15.65 | 110.0 | 113.3 | 103.4 | 108.9 | 106.5 | +2.3 |
| d&e14(80%) | 8.02 | 210.5 | 12.39 | 82.7 | 87.1 | 72.9 | 80.9 | 79.3 | +2.0 |
| ed 564(80%) | 10.76 | 232.6 | 14.56 | 112.9 | 99.4 | 92.7 | 101.6 | 90.9 | +11.8 |
| 543(60%) | 5.81 | 156.2 | 9.62 | 59.0 | 59.6 | 50.6 | 56.4 | 62.2 | -10.3 |
| 492(60%) | 5.81 | 182.6 | 9.00 | 59.0 | 72.5 | 46.1 | 59.2 | 67.7 | -14.3 |
| 527(60%) | 8.17 | 198.5 | 12.50 | 84.3 | 80.8 | 73.8 | 79.6 | 82.4 | -3.5 |

1. The listed data has been normalized to 1680V to use the MTR calibration.
2. The AmLi background rates of 187 cps for the front Singles and 1.9 cps for the back Singles has been subtracted from the data.
4. The measured rates were increased by a factor of 1.05 because of the fuel element positioning in the sample channel and the burnup profile.

## 8. Results – Passive Neutron and Gamma Measurements

The HIFAR fuel has UAlx alloy fuel for the HEU cases and Silicide ($U_3Si_2$-Al) fuel for the LEU. The active mode assay for the $^{235}U$ is independent of the fuel composition but the passive mode neutron rate has a dependence because Si and Al have (a,n) yields that are different.

The gamma results need to be corrected for cooling time before the relative burnup can be determined from the gross gamma measurement. Most of the Pu isotopes have long decay constants and no decay correction is needed, but for Pu-241, the neutron yields will be cooling-time dependent. The burnup codes were used to determine if the curium isotopes contribute to the neutron rate. The initial results indicate a negligible Cm contribution.

The passive gamma-ray activity was measured using the ion chamber for the fuel element in the reference position (~ midway between the center and the top end of the fuel zone). For cooling times of less than ~3 y, the gamma activity is primarily a function of the cooling time and the reactor power level. For long cooling times the decay of Cs-137 becomes the dominant activity and the 37 year half-life provides a relative verification of the declared burnup. All of the gamma data is consistent with the declared cooling times and burnups. The verification of the declared burnup is obtained with better accuracy using the active mode neutron measurement of the residual $^{235}U$.

The passive measurement counting rates were extracted from the MiniGRAND (MG) data and are shown in Table 6. The data has been corrected for the HV shift for the high dose fuel.

**Table 6. Passive measurement results – neutron and gamma**

| Assembly (enrichment) | Passive Single (4 tubes) cps | Passive Singles (2 tubes) cps | Ratio | Gamma MG units |
|---|---|---|---|---|
| 382(19.9%) | 550.6 | 26.0 | 21 | 147 |
| 727(80%) | 85.4 | 4.8 | 18 | 4.90 |
| ed14(80%) | 29.6 | 1.7 | 18 | 4.12 |
| 564(80%) | 107.3 | 6.1 | 18 | 6.59 |
| 543(60%) | 1984.3 | 100.0 | 20 | 84 |
| 492(60%) | 1887.9 | 94.2 | 20 | 50 |
| 527(60%) | 1165.4 | 66.6 | 18 | 39 |

The ratio of the neutron counting rate from the front 4 tubes to the counting rate from the rear 2 tubes is almost constant as would be expected. This is a confirmation that the tubes are operating correctly. The value from the front 4 tubes will be used because of their better statistical precision.

Figure 9 shows a plot of the measured neutron rate versus the measured gamma rate, and the LEU fuel element falls far to the right on the plot. The neutron Singles emission alone is not sufficient to distinguish the LEU element from the HEU elements. The measured rate of the LEU assembly is between that of the 80% and 60% elements. The data represents a wide range of important parameters: burnup, cooling time, fuel composition.

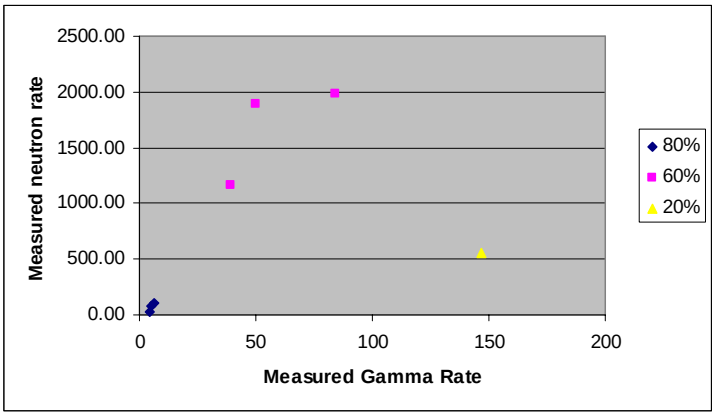



*Figure. 9.* ***Declared U-235 mass versus net Singles in the back He-3 detectors.***

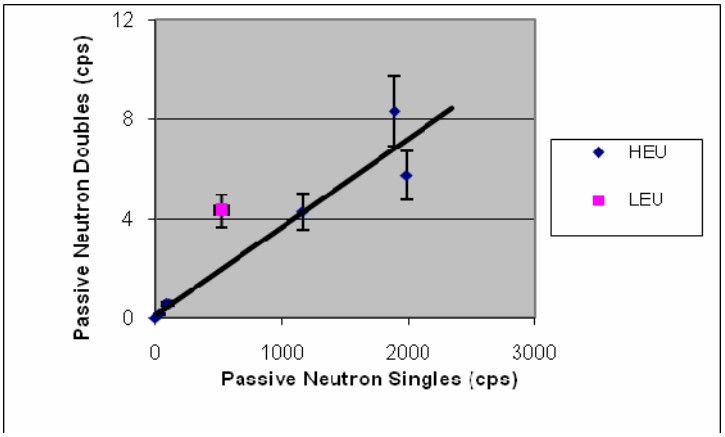



***Figure. 10.*** *Experimental values of neutron Doubles rate versus measured Singles rate.*

The LEU point has a much lower neutron to gamma ratio than the HEU because of the short cooling time and silicide fuel composition, and the data does not unambiguously distinguish

HEU from LEU. However, the data does show neutron and gamma rates that are consistent with the declarations.

### 8.1. LEU Versus HEU via Passive Neutron Doubles and Singles Measurements

The passive neutron emission from the spent fuel comes primarily from the spontaneous fission (S,F) and alpha,n activity from the decay of $^{238}$Pu, Pu-240, and curium. During the reactor irradiation, the multiple neutron captures in the $^{235}$U produce the $^{238}$Pu,; whereas, the multiple neutron captures in the U-238 produce the Pu-240 and curium resulting in a higher S,F rate. Because the ratio of U-238 to $^{235}$U is higher for LEU than HEU, the S,F/alpha,n ratio will be higher for LEU than HEU. This is true for fuel with similar burnups and the same alloy composition. If the fuel has different compositions and burnups, corrections to data can be made based on calculations. Fortunately, the ratio of S,F and alpha,n neutron yields are different by a factor of ~ 38 for Pu-240 and $^{238}$Pu,, and the Doubles to Singles ratio will be different for LEU compared to HEU. The U-238 neutron capture reactions can also lead to the production of curium that has more S,F neutrons than alpha,n neutrons. Table 7 lists the S,F and alpha,n neutron rates for oxides for the HEU and LEU neutron sources [7]. The alpha,n rates will increase for an UAlx alloy (HEU) compared with oxides and silicides.

**Table 7. Neutron source terms for S,F and alpha,n reactions**

| Isotope | Dominant Origin | S,F (n/s.g) | Alpha,n (n/s.g oxide) | Ratio S,F/alpha,n |
|---|---|---|---|---|
| $^{238}$Pu | HEU | 2.59E+03 | 1.34E+04 | 0.193 |
| $^{240}$Pu | LEU | 1.02E+03 | 1.41E+02 | 7.23 |
| $^{242}$Cm | LEU | 2.10E+07 | 3.76E+06 | 5.59 |
| $^{244}$Cm | LEU | 1.08E+07 | 7.73E+04 | 140 |

The Doubles rate is more sensitive to the S,F neutrons because they are born in coincidence, so the Doubles versus Singles plot should differentiate the HEU from the LEU. Of course, the HEU contains initial U-238 and the LEU contains initial $^{235}$U (19.9%). The initial $^{235}$U mass in the LEU is typically similar to HEU as illustrated in Table 3; however, the $^{238}$U mass is higher by a factor of 3-20 for the LEU assemblies.

The use of Doubles versus Singles to differentiate LEU from HEU will be diluted from the factors indicated in Table 6 because both $^{235}$U and $^{238}$U are present in the LEU and HEU. However, there should still be a significant difference between LEU and HEU in the Doubles and Singles data. Figure 10 shows the measured Doubles rates versus Singles rates for the seven HIFAR fuel elements and we see that the LEU Doubles rate is above the line fit to the HEU by more than a factor of two. To a first approximation, one neutron reaction can produce Pu-239, two neutron events (n captures in $^{238}$U) are needed to produce Pu-240; however, for $^{235}$U, three neutron events are needed to produce $^{238}$Pu, (n captures in $^{235}$U). Thus, the buildup of $^{238}$Pu, starts out slower with burnup than Pu-240.

Three of the HEU (80% $^{235}$U) elements for the HIFAR data set have a relatively low burnup (~ 30%), and they have a low neutron Singles rate because the buildup of $^{238}$Pu, is in the early

stages relative to the other elements. In Fig. 10 we see that the corresponding three data points near the origin are not on the same curve as the three HEU elements with high burnup (~ 60%).

## 9. SUMMARY

The calibration of the AEFC for MTR-type fuel elements was completed at LANL prior to shipping the detector to Australia. The efficiency was 4.6% for counting Cf-252 neutrons at the midpoint of the sample position; the signal-to-background (AmLi source) ratio was 18 for the pair of back $^{3}$He tubes counting in the totals mode for the fresh MTR fuel element. For the HIFAR fuel elements, the statistical precision of ~ 2% for Doubles was obtained in a 10 min. measurement.

The equipment setup and measurements were performed over a 3 day period (Nov 10-14) at the HIFAR reactor. The AEFC system uses IAEA approved detectors, electronics, and software for verification purposes. Both active and passive mode measurements were performed for seven spent fuel assemblies that included HEU and LEU enrichments. The residual $^{235}$U mass was measured in the active mode using the AmLi neutron source for the interrogation.

Some of the key findings are:

- The active mode neutron measurements gave very good agreement with the declared residual $^{235}$U mass. The average measured $^{235}$U mass versus the declared mass gave a relative standard deviation of 5% for the 7 elements (Table 9).
- The plot of the ratio of neutron to gamma activity (Fig. 9) provides a clear distinction of HEU from LEU for the HIFAR fuel. However, much of the LEU versus HEU distinction for the HIFAR fuel is the result of the higher alpha,n yield from the UAlx alloy for the HEU versus the silicide for the LEU. More extensive burnup and source yield calculations will be needed to extend this conclusion to other types of research reactor fuel.
- The neutron Doubles rate versus the Singles rate (Fig. 10) shows a significant separation of the LEU from the HEU and indicates that the Doubles/Singles ratio is higher for LEU than for HEU for the HIFAR fuel. However, for the generic case of any burnup and cooling time, a more complete study of the parameter space will be needed to conclude that LEU can be distinguished from HEU.
- The ion chamber shielding needs to be redesigned to obtain an accurate γ burnup profile. The profile measured in the present work is much too broad because of end leakage of the gammas from the fuel element that extended beyond the lead shielding. The shielding of the He-3 tubes needs to be increased to extend the measurement capability to spent fuel with a cooling time of less than five years.
- It was observed on the first day that it was not possible to lower the AmLi source down the entire length of the Tygon tube when it was under 5 m of water due to a compression of the tube. This problem was resolved by replacing the tube with two 3 m lengths of PVC piping. LANL will redesign the AEFC to use an alternative to the Tygon tubing.
- The burnup of the measured fuel elements varied from around 28% to 64%, and the cooling time varied from ~ 1 year to 40 years. The γ yield was dominated by the cooling time.

- In general, the passive neutron and γ yields from spent fuel are a function of the initial enrichment, the burnup, the cooling time, multiplication, and the composition of the uranium deposit. The AEFC active neutron interrogation of the spent fuel provides an unambiguous verification of the burnup and the residual U-235 mass that is independent of the reactor parameters. However, the passive neutron and γ determination of the initial enrichment is a more complex combination of the reactor and fuel parameters. Burnup, MCNPX, and Sources code calculations are required to better simulate the passive signals. The present measurements verified that the declared fuel parameters for HIFAR fuel were consistent with the measured data. Further parametric study is required to determine the conditions under which LEU and HEU can be distinguished in the general case.

## 10. Acknowledgements

We would like to acknowledge the Department of Energy National Nuclear Security Administration's Office of Dismantlement and Transparency (NA-241) for their support in the design and development of the AEFC, and the Program of Technical Assistance to the IAEA (POTAS) for the field work support and calibration activities. The ANSTO staff at the HIFAR reactor were very helpful in supplying the fuel movements, logistical support, and health physics monitoring and oversight.

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# Performance testing of the device for safeguards verification of the damaged fuel at Paks NPP

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***Abstract:***

*Test measurements have been performed with the equipment which is planned to be applied for determining the mass of nuclear material in a mixture of damaged spent fuel packed into containers at Paks NPP in Hungary. The tested equipment consisted of a high-resolution gamma spectrometer, and an underwater device containing two fission chambers for neutron counting and a CdTe detector for medium resolution gamma spectroscopy.*

*Ten spent fuel assemblies were scanned along their length, each from at least three of its sides, using the above equipment. A total of about 700 gamma spectra, as well as about 350 neutron pulse amplitude spectra were taken. Based on the gamma and neutron profiles and on the burn-up values available from depletion calculations, correlations have been established between the intensity of the 661 keV peak of $^{137}$Cs, the $^{134}$Cs/$^{137}$Cs ratio as well as the neutron count rate measured at the central position along the assemblies' height and the known burn-up at the middle of the assemblies.*

*Determining the mass of nuclear material is based on the principle that the nuclear mass is equal to the product of the concentration of the nuclear material and of the total mass of fuel in the observed volume. The mass of nuclear material in the whole assembly or container is then determined by integrating along the length of the examined item. The concentration of the nuclear material, in particular of U-235, total U and total Pu, is calculated using correlations between the concentrations and the burn-up values determined from the gamma-spectrometric data. In addition, a new model is presented for evaluating the total mass of spent fuel in the observed volume from a burn-up dependent relationship between the neutron count rate and the fuel mass.*

**Keywords:** spent fuel; nuclear-material content; NDA

## 1. Introduction

The purpose of this paper is to present the results of the test measurements performed with the equipment which is to be used for determining the nuclear material content of the containers containing damaged VVER-440 reactor fuel at Paks NPP in Hungary.

At Paks NPP about 60 containers have been loaded with the fuel damaged in the 2003 incident at the NPP. In the incident on April 10, 2003 in unit 2 of Paks NPP 30 fuel assemblies were damaged and became unusable [1], [2], [3], [4]. The incident was rated level 3 on the International Nuclear Event Scale (INES). The elements of the damaged fuel assemblies have been repackaged into closed containers, which will be kept in the spent-fuel pond until further action [5].

The damaged-fuel containers will be under IAEA and EURATOM safeguards. The nuclear-material content of each individual container will be determined and declared, in line with the safeguards requirements. The amounts of $^{235}$U, total uranium and plutonium will have to be determined and declared for each container individually, in compliance with IAEA requirements.

The first version of the equipment to be used for determining the nuclear-material content of the containers consisted of high-resolution gamma spectrometer placed behind a collimator built into the wall of the service pit beside the spent fuel pond and of an underwater device containing two fission chambers for neutron counting and a Cd(Zn)Te detector. The design of the underwater device is similar to the “Enhanced Fork Detector” [6], [7], [8] and, to some extent, to the “SMOPY” device [9]. It was tested with 10 regular spent fuel assemblies in April 2006. In addition, gamma spectra of the same 10 assemblies were also taken by a high-resolution gamma spectrometer. The measurement campaign to determine the nuclear material content of the damaged fuel is scheduled for November and December 2007.

## 2. Status of the damaged fuel

In the incident of April 2003, fuel rods broke and pellets and parts of the cladding fell out from the assemblies and piled up at the bottom of the cleaning tank [10], [11], [12]. The initial inventory of nuclear material in the assemblies (i.e. before the incident) is known and documented. In the incident, however, pellets from different types of assemblies of different burn-up and irradiation history got mixed together at the bottom of the tank. Now they cannot be separated nor identified.

The damaged fuel has been re-packaged into closed containers by the end of March 2007. There are three types of containers used for the remnants of the damaged assemblies [13]. Two types are used for nuclear material, and one type for the non-nuclear construction elements (i.e. assembly heads and tails). The dimensions of the containers are similar to those of the fuel assemblies, so that the loaded containers can be placed into the lattice in the spent-fuel pond.

During the recovery process of the damaged fuel, weight measurements have been performed for each loaded container. Hence, an upper bound for the mass of nuclear material can be estimated for each container based on the total mass of the material loaded into the containers.

## 3. Experimental setup

**Fig. 1.** The fork-shaped device.

### 3.1. High-resolution gamma spectrometry

A HPGe detector was placed behind the collimator built into the concrete wall of the service pit of the reactor block. The investigated spent fuel assembly was moved up ad down under water in the service pit in front of the collimator, by the refuelling machine. The width of the collimator opening was ~20cm, while its height was ~1 cm, making it possible to collect gamma spectrometric information with a relatively high spatial precision. During the test measurements in April 2006 gamma spectra were taken from 3 sides of 9 assemblies and from 5 sides of one assembly. The HPGe detector was an ORTEC SGD GEM 3615 connected by a ~30 m long cable to an ORTEC DART multi-channel analyzer controlled by the Gammavision software installed on a laptop computer. The laptop computer was connected through a local area network to another laptop PC, placed in the reactor hall, close to the control boot of the refuelling machine and the spectrum acquisition (e.g. acquisition start, acquisitions stop, spectrum saving) was controlled from the reactor hall over the local area network.

### 3.2. Neutron counting and medium resolution gamma spectrometry

The detector unit is a fork-shaped box made of stainless steel, which contains a KNK15 type fission chamber in each of its "prongs" (see Fig. 1) and a CdTe detector (RITEC, 20 $mm^3$) with a lead collimator in its "handle"[1]. Unlike the "traditional" FORK detector, in the present device the fission chambers are placed parallel to the fuel assembly. The round-shaped opening of the collimator was such that the CdTe detector viewed a 20 cm high portion of the assembly. The investigated spent fuel assembly (or a container with the damaged fuel) is placed between the prongs of the fork and moved up and down with the refuelling machine. The detector unit is watertight, but its back can be removed for installing the detectors into the box and for eventual servicing purposes.

This device can be used either in the spent fuel pond or in the service pit. During the measurement campaign in 2006 the device was used in the spent fuel pond. It was placed into the pond fixed to a frame having three conical-shaped legs made of lead, which fit into the holes of the lattice in the pond. The amplified detector signals were connected to multi-channel analyzers, placed beside the spent fuel pond in the reactor hall, and controlled by the ORTEC Gammavision software.

## 4. Measurements

In the test measurements 10 spent fuel assemblies having burn-up from ~10 GWd/tU to ~ 40 GWd/tU have been investigated. The assemblies spent 1 to 4 cycles in the reactor, and the time elapsed from their last discharge from the reactor varied between 1.5 and 3.5 years. There were two assemblies with initial $^{235}U$ enrichment of 1.6 %, one of 2.4 % and 7 of 3.6 % or 3.8 %. With the HPGe detector measurements were performed at 11 height positions from 3 sides of each assembly (and from 5 sides of one assembly). With the device containing the CdTe detector and the fission chambers measurements at 10 height positions were performed, also from 3 (respectively 5) sides. A total of about 350 amplitude spectra of the fission chamber signals and 700 gamma spectra were recorded. The measurement campaign lasted about one week, in three eight-hour shifts, 24 hours a day.

### 4.1. Gamma measurements

Gamma spectra taken with the HPGe and the CdTe detector are shown in Fig. 2.

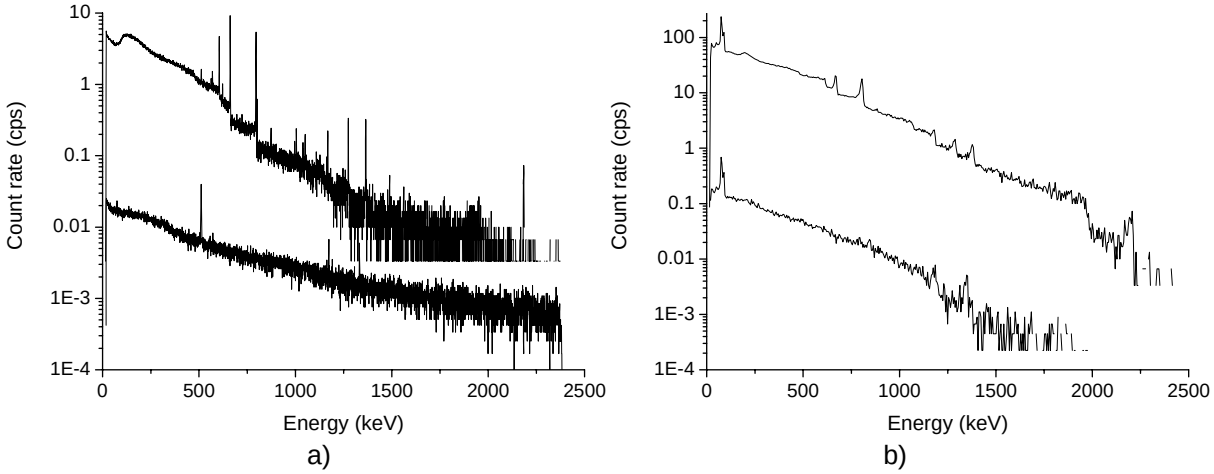
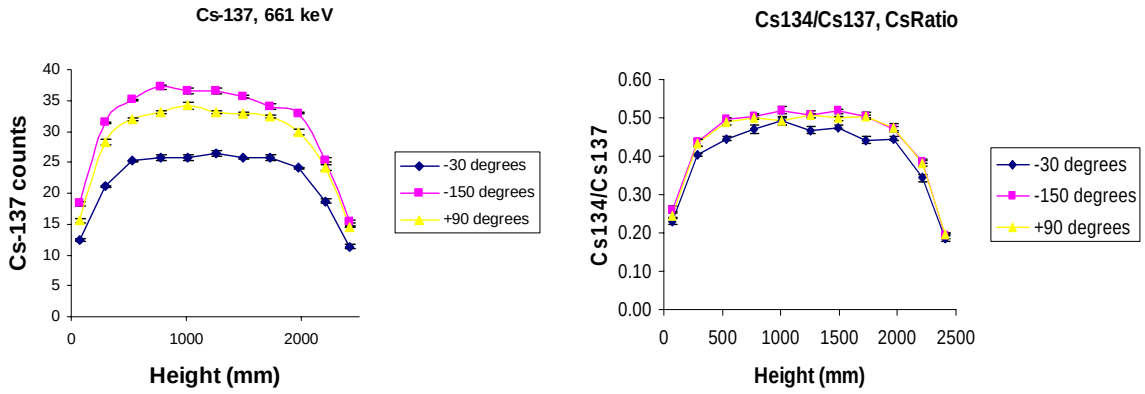



**Fig. 2.** Gamma spectra of the background and of a spent fuel assembly with average burn-up of 29.766 GWd/tU: a) HPGe detector; b) CdTe detector

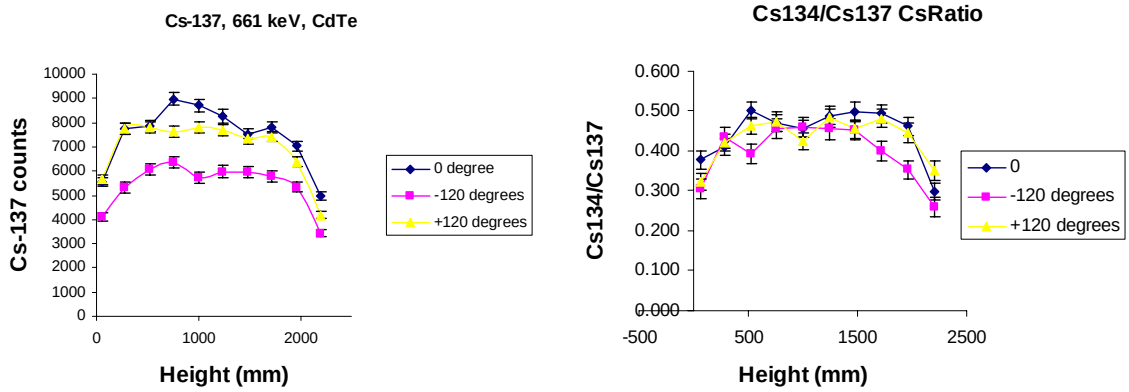
[1] In the test measurements in April 2006 a pair of bubble detectors was also placed into the box, with on-line acoustic readout of the neutron count rate and with remote-controlled bubble resetting mechanism. Because of radiation damage to the detectors and to the optical cables used for signal transfer, their use has been abandoned in subsequent measurement campaigns. Instead, now a pair of Si-diodes is placed into the device for gross-gamma counting.

From the recorded gamma spectra the peaks of $^{134}$Cs at 605 keV and 796 keV and the $^{137}$Cs peak at 661 keV were evaluated. The $^{134}$Cs/$^{137}$Cs activity ratio was calculated using the CsRatio software. The $^{137}$Cs profile and the $^{134}$Cs/$^{137}$Cs activity ratio profile of an assembly, taken by the HPGe detector from 3 sides of the assembly are shown in Fig. 3. In addition, Fig. 4 shows the analogous profiles of the same assembly, taken by the CdTe detector.




a) b)

**Fig. 3.** a) HPGe profiles of an assembly with average burn-up of 29.766 GWd/tU: a) $^{137}$Cs profile based on the 661 keV gamma line; b) $^{134}$Cs/$^{137}$Cs activity ratio profile based on the peaks of $^{134}$Cs at 605 keV and 796 keV and the $^{137}$Cs peak at 661 keV.




a) b)

**Fig. 4.** The same as Fig. 3 but taken with the CdTe detector

It can be seen from the gamma profiles that the count rates of the 661 keV line and the Cs activity ratio are different for the three sides of the assembly. This effect was observed in 8 assemblies out of 10 and the difference between the least and the most intense side is about 5-30 %, depending on the assembly. The possible reason of this asymmetry is the asymmetric burn-up of the assemblies [8] depending on the gradient of the neutron flux in the reactor core. Another cause might be a possible bad reproducibility of the measurements (e.g bad positioning accuracy). These speculations, however, have not yet been quantitatively assessed.

Based on the gamma measurements averaged over three sides of the assemblies and on the burn-up values available from depletion calculations, correlations have been established between the burn-up at the middle of the assemblies and the intensity of the 661 keV peak of $^{137}$Cs and the $^{134}$Cs/$^{137}$Cs activity ratio measured at the central position along the assemblies' height. For establishing the correlations only the assemblies with initial enrichment 3.6 % and 3.8 % were used.

In particular, it is assumed that the activity of $^{137}$Cs is proportional to the burn-up of the nuclear material, while the activity of $^{134}$Cs is proportional to the square of the burn-up. This implies that the count rate of the 661 keV line of $^{137}$Cs and also the count-rate ratio $^{134}$Cs/$^{137}$Cs should be proportional to the burn-up. The count-rate ratio, however, is less sensitive to the changes of the measurement

geometry. In particular, based on the count rates at the middle of the spent fuel assemblies averaged for three sides of the assemblies, the following empirical relationships hold:

$$BU = k\frac{A(Cs134)}{A(Cs137)}\exp[(\lambda_{Cs134} - \lambda_{Cs137})t] \ , \qquad (1)$$

$$BU = k_{137} I(Cs137)\exp[\lambda_{Cs137} t]$$

where *BU* is the burn-up, *A(Cs134)/A(Cs137)* is the $^{134}$Cs/$^{137}$Cs activity ratio *I(Cs137)* is the count rate of $^{137}$Cs, *t* is the cooling time, the $\lambda$-s are the corresponding decay constants and we have obtained by measurement that in the applied measurement setup $k$=1/0.0855=11.7 GWd/tU and $k_{137}$=1/1.0553=0.948 GWd/tU/cps for the HPGe detector.

The burn-up values of the investigated spent fuel assemblies were supplied by the Paks NPP, based on depletion calculations performed at 20 nodes along the assemblies, using the C-PORCA software developed at the Paks NPP. The concentrations of the main fission products were also calculated at 20 nodes along each assembly, using the SCALE-4.4a software suite. A comparison of the calculated burn-up profile and the one obtained from the HPGe measurements is shown in Fig. 5 for the same assembly as in Fig. 3 and Fig. 4. A third degree spline function was fitted to the calculated data points.

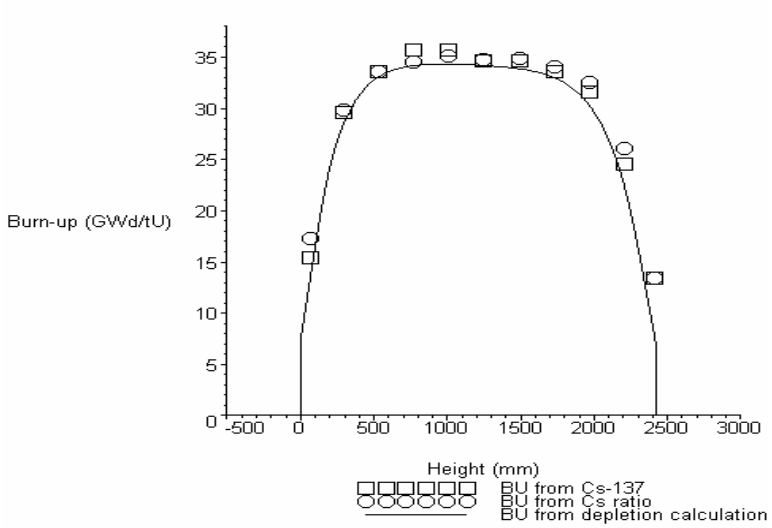



**Fig. 5.** The calculated and the measured burn-up profile for an assembly with average burn-up of 29.766 GWd/tU. The circles correspond to the burn-up obtained from the measured $^{134}$Cs/$^{137}$Cs ratio, while the line corresponds to the depletion calculations.

### 4.2. Neutron measurements

The amplitude spectra of the fission chamber signals were recorded through the Mini MCA 166, with the ADC set to 512 channels. The amplitude spectra were integrated above the 150$^{th}$ channel to obtain the neutron counts. A “neutron spectrum” and the corresponding background are shown in Fig. 6. Note that the beginning of the amplitude spectrum of the measurement with the spent fuel assembly is shifted to the right with respect to the background. This was due to the influence of the high gamma dose rate. Nevertheless, this effect does not influence the integral above the 150$^{th}$ channel, which is therefore proportional to the neutron emission rate. The “peak” at the end of the spectrum is merely due to the saturation of the amplifier. The neutron profile of the same assembly as above is shown in Fig. 7. Similarly to the gamma measurements, the neutron count rates are different for the three sides of most of the assemblies.

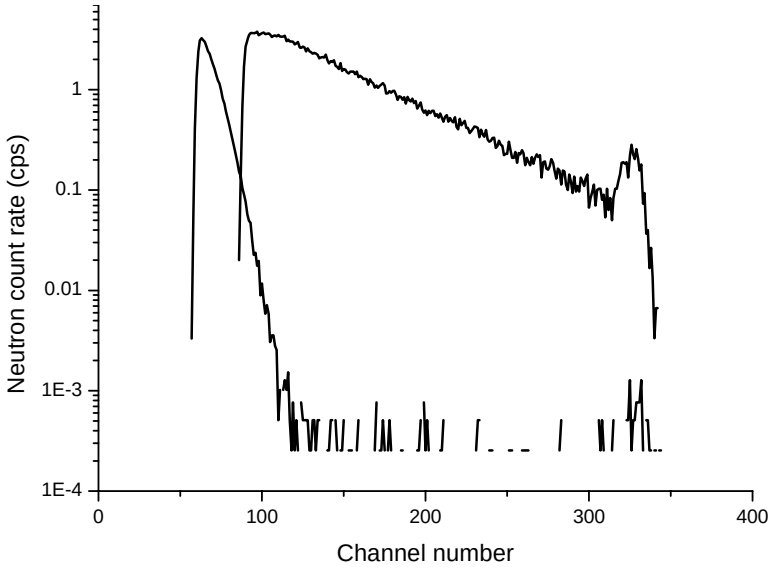



**Fig. 6.** Amplitude spectra of the fission chambers. The background and a neutron measurement at the middle of an assembly with average burn-up of 29.766 GWd/tU are shown.

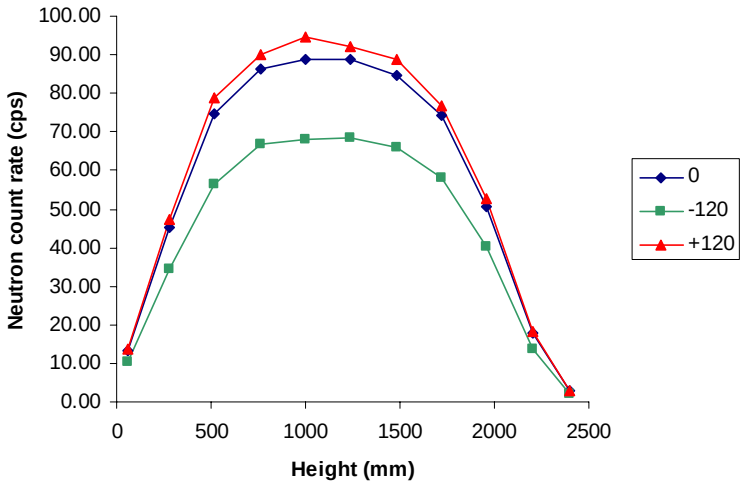



**Fig. 7.** Neutron profile of an assembly with average burn-up of 29.766 GWd/tU.

The neutron count rate measured at the middle height position of each assembly, averaged for three sides of the assemblies was correlated to the burn-up values available from the above mentioned depletion calculations. Assuming a power law between the burn-up, $BU_{middle}$, and the neutron count rate corrected for $^{244}$Cm decay, $N_{middle}$, we obtained that around the middle height position of the assemblies the following relationship holds

$$N_{middle} = \alpha \frac{M}{L} BU_{middle}^{\beta}, \qquad (2)$$

where $M$ is the total mass of nuclear material in the assembly and $L$ is its length (height). If $N_{middle}$ is given in cps and $BU_{middle}$ is given in GWd/tU, then $\alpha M/L \approx 1.36 \times 10^{-5}$ and $\beta \approx 4{,}4$. For establishing this correlation only the assemblies with initial enrichment 3.6 % and 3.8 % were used. Furthermore, for VVER-440 assemblies the dependence of the parameter $\alpha$ on neutron multiplication can be neglected. The neutron count rates measured at the middle of the assemblies are shown versus burn-up on Fig. 8.

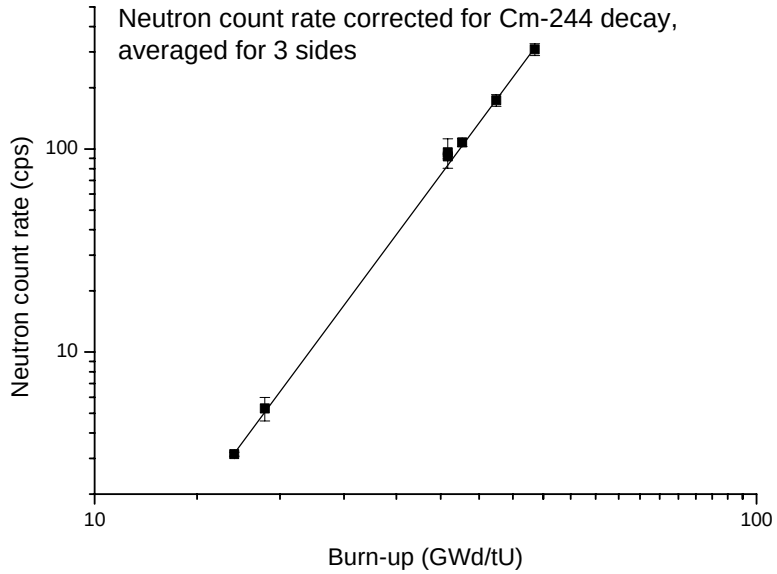



**Fig. 8.** Neutron count rates measured at the middle of the assemblies

## 5. Determining the mass of nuclear material in spent fuel assemblies

Determining the mass of nuclear material in a spent fuel assembly (and, ultimately, in a container with the damaged fuel) is based on the principle that the mass of nuclear material, *dm*, is proportional to the product of the concentration, $\rho$, of that particular type of nuclear material and of the total spent fuel mass, $dm_{fuel}$, in the observed volume:

$$dm = \rho dm_{fuel} . \tag{3}$$

From this relationship one obtains that the mass of nuclear material, *m*, in the entire assembly (or container) can be obtained by integration along the full length of the assembly (or container):

$$m = \int_0^L \rho \frac{dm_{fuel}}{dx_{fuel}} dx_{fuel} \quad , \tag{4}$$

where *x* is the coordinate along the length (height) of the assembly and *L* is the total length of the fuel within the assembly. For regular spent fuel assemblies the mass of spent fuel per unit length is constant, that is, $dm_{fuel}/dx_{fuel} = const = M/L$, where *M* is the total mass of spent fuel in an assembly, holds for regular, un-damaged spent fuel assemblies. For any container with the damaged fuel, however, $dm_{fuel}/dx_{fuel}$ strongly depends on the position along the container's length.

### 5.1. Determining the concentration of nuclear material along the assemblies' length

By "concentration" of nuclear material here we mean the ratio of the mass of $^{235}U$, $^{238}U$ and total Pu to the total mass of the spent fuel (which is practically the same as the initial mass of uranium). The concentration of nuclear material in the investigated spent fuel assemblies is determined from the correlations between the burn-up and the amounts of $^{235}U$, $^{238}U$ and total Pu remaining in the spent fuel. We used the correlations obtained by depletion calculation codes for VVER-440 assemblies given in reference [14]. In the burn-up range of interest for us, the concentration of the nuclear material remaining in the spent fuel is calculated at each measurement position along the assemblies' length using the formulas

$$\begin{aligned} \rho(U_{total}) &= k_0(U_{total}) + k_1(U_{total})BU \\ \rho(U235) &= k_0(U235) + k_1(U235)BU \\ \rho(Pu) &= k_1(Pu)BU + k_2(Pu)BU^2 \end{aligned} \tag{5}$$

where the *k*-s are calibration constants. Using the data from [14] we obtained $k_0(U_{total})\approx 0.9653$, $k_1(U_{total})\approx -7.347\text{e-}4$, $k_0(U235)\approx 0.03528$, $k_1(U235)\approx -0.00107$, $k_2(U235)\approx 9.25308\text{e-}6$, $k_0(Pu)\approx 2.1677\text{e-}4$, $k_1(Pu)\approx 4.3478\text{e-}4$ and $k_2(Pu)\approx -4.9580\text{e-}6$. The burn-up, *BU*, was determined in each measurement position along the assemblies' length from the Cs activity ratio. The concentration profiles are analogous to the measured burn-up profile shown in Fig. 5.

### 5.2. Determining the total mass of spent fuel per unit length of the assemblies

In this paper we present a method based on neutron counting for determining the total mass of the spent fuel. By generalizing eq. (2), we assumed that the number of neutrons, $dN(x_{det})$, emitted by spent fuel of mass $dm_{fuel}$ and burn-up $BU(x_{fuel})$ at position $x_{fuel}$ and registered by the infinitesimal volume of the detector at position $x_{det}$ (see Fig. 9) may be given using the relationship

$$dN(x_{det}) = \alpha f(x_{\det}, x_{fuel}, \Delta y)dx_{det}dm_{fuel}[BU(x_{fuel})]^{\beta} \quad , \tag{6}$$

where $\alpha$ and $\beta$ have been given above while $f(x_{det}, x_{fuel}, \Delta y)$ is a function describing the geometrical efficiency of the detector together with the neutron attenuation between the observed piece of spent fuel and the infinitesimal volume of the detector.

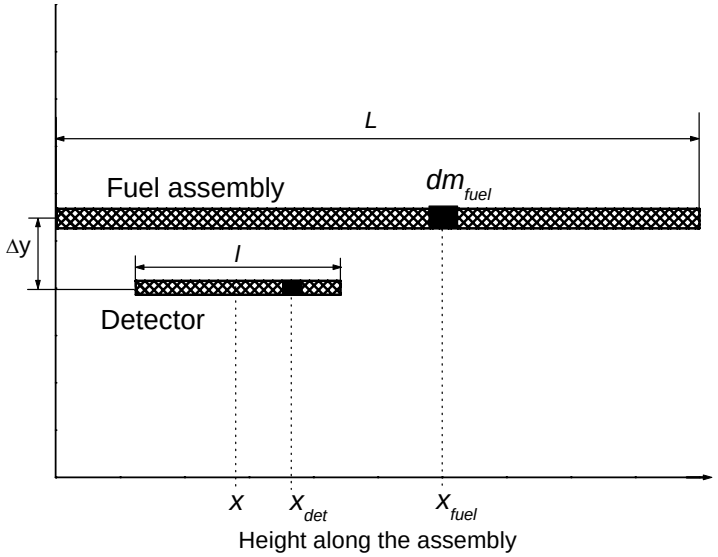



**Fig. 9.** Explanation of the coordinates used eqs. (6) and (7). (The ratios on the drawing do not correspond to the real ratios.)

This implies that the (cooling-time corrected) number of neutrons registered by the detector at position *x* can be given as

$$N(x) = \alpha \int_{-\infty}^{\infty} \left[ \int_{x-l/2}^{x+l/2} f(x_{\det}, x_{fuel}, \Delta y)dx_{\det} \right] \frac{dm_{fuel}}{dx_{fuel}}(x_{fuel})[(BU(x_{fuel})]^{\beta} dx_{fuel} \tag{7}$$

where *l* is the length of the detector. Note that by using a one dimensional model of the fuel assembly we have implicitly neglected the neutron self-attenuation within the fuel.

The form of the function $f(x_{det}, x_{fuel}, \Delta y)$ is not known. However, the normalization condition requires that

$$\int_{-\infty}^{\infty} \int_{x-l/2}^{x+l/2} f(x_{\det}, x_{fuel}, \Delta y)dx_{\det}dx_{fuel} = 1 \tag{8}$$

for any *x* and any $\Delta y$. In addition, by recording the geometrical response function of the fission chambers using an Am-Be neutron source under laboratory conditions we have established that it is reasonable to assume that

$$\varepsilon(x, x_{fuel}, w) := \int_{x-l/2}^{x+l/2} f(x_{\det}, x_{fuel}, \Delta y)\, dx_{\det} \approx \frac{e^{2\left(\frac{x-x_{fuel}}{w}\right)^2}}{\int_{-\infty}^{\infty} e^{2\left(\frac{x-x_{fuel}}{w}\right)^2} dx} \qquad (9)$$

That is, in our case, numerically a Gaussian function equally well describes the data points as a more complicated function would, therefore, for the sake of simplicity and faster computation of the result, we chose to use a Gaussian function with an adequately adjusted parameter *w*. The Gaussian width *w* in eq. (9) depends on the detector size (*l*) and the distance of the detector from the assembly (*Δy*). Since the detector size and the distance of the detector from the assembly are constant, *w* is also constant for all the measurements with the same device. We determined *w* from the geometrical response function of the fission chambers recorded under laboratory conditions and also by fitting the neutron profile calculated from eq. (7) to the experimentally obtained neutron count rates of the spent fuel assemblies. In this way we obtained *w*=270 ± 60 mm. The boundaries of integration (-∞,∞) in eq. (7) can be replaced by the interval [0,L], since outside this interval both $dm_{fuel}/dx_{fuel}$ and $BU(x_{fuel})$ are identically zero. (Note, however, that, in principle, *N(x)>0* everywhere!) The function $BU(x_{fuel})$, giving the burn-up profile along an assembly, is a third degree spline function fitted to the measured data points.

Note that at the middle of the assemblies the burn-up can be assumed to be constant along a portion of the assembly which is larger than the length of the neutron detectors. Furthermore, for regular spent fuel assemblies $dm_{fuel}/dx_{fuel}$ is also constant. Therefore, by inserting *BU=const* and $dm_{fuel}/dx_{fuel} = M/L$ into eq. (7) one obtains exactly eq. (2).

The mass of spent fuel per unit length, $dm_{fuel}/dx_{fuel}$, which is used in eq. (4) for determining the mass of nuclear material, can be calculated from eq. (7), which is a Fredholm-type integral equation of the first kind. In principle, this type of integral equations can be solved by inverse Fourier transformations, provided that all the function appearing in the equation satisfy suitable continuity etc. conditions. When these functions, however, are only known in a finite number of points (such as in our case when the values of *N(x)* are only known in those points where a measurement has been done), appropriate numerical methods should be applied in order to determine the function $dm_{fuel}/dx_{fuel}$ from eq. (7). A wealth of methods for numerically solving Fredholm-type integral equations of the first kind exists in the mathematical literature, each fitted to a specific purpose. One of these methods will have to be used for determining $dm_{fuel}/dx_{fuel}$ in the case of the containers containing the damaged fuel from Paks NPP. In the case of regular spent fuel assemblies, however, the function $dm_{fuel}/dx_{fuel}$ is constant along the assembly; therefore it can be taken out from the integral. Thus, from eq. (7) one obtains for a *regular spent fuel assembly*

$$\frac{dm_{fuel}}{dx_{fuel}} = \frac{N(x)}{\int_{-\infty}^{\infty} \alpha\varepsilon(x, x_{fuel}, w)[(BU(x_{fuel})]^{\beta}\, dx_{fuel}} = \frac{M}{L} = const. \qquad (10)$$

Since for regular spent fuel assemblies *M* and *L* are known, the above expression can be used for assessing the performance of the applied equipment for determining the mass of nuclear material in spent fuel. Here we wish to emphasize again, that in the case of the damaged fuel, which is a mixture of pieces of various burn-up and $dm_{fuel}/dx_{fuel}$ varies along the container, the above expression cannot be used, and eq. (7) has to be solved by other methods.

In Fig. 10 the measured mass of spent fuel per unit mass is shown for two chosen test assemblies, together with the known value of *M/L*. This figure well illustrates the deviations of the measured $dm_{fuel}/dx_{fuel}$ from constancy and the error associated with determining the mass of spent fuel, and consequently of the mass of nuclear material in an assembly.

Inserting the measured function $dm_{fuel}/dx_{fuel}$ from eq.(10) and the concentrations from eq. (5) to eq. (4) one obtains the mass of nuclear material. In particular, for the two selected assemblies from Fig. 10 the values for the mass of nuclear material are given in Table 1.

| | Burn-up=35.42 GWd/tU | | | Burn-up=29.77 GWd/tU | | |
|---|---|---|---|---|---|---|
| | Measured mass (kg) (burn-up from $^{134}Cs/^{137}Cs$) | Measured mass (kg) (burn-up from $^{137}Cs$) | Mass from depletion calculations* (kg) | Measured mass (kg) (burn-up from $^{134}Cs/^{137}Cs$) | Measured mass (kg) (burn-up from $^{137}Cs$) | Mass from depletion calculations* (kg) |
| $^{235}U$ | 1.05 | 1.07 | 1.30 | 1.33 | 1.19 | 1.61 |
| Total U | 109.73 | 109.63 | 114.32 | 111.86 | 101.86 | 115.24 |
| Total Pu | 1.09 | 1.08 | 1.19 | 1.04 | 0.96 | 1.10 |
| Total SF | 116.90 | 116.77 | 119.80** | 118.68 | 108.13 | 119.94** |

**Table 1.** The mass of nuclear material and of the total mass of spent fuel for two selected assemblies (*As provided by the NPP. **Initial U-mass.)

As it can be seen from Table 1, the absolute bias between the measured mass of spent fuel (SF) in a regular spent fuel assembly and the initial mass of fuel (i.e. total initial mass of U) is less than 10 %. For the nuclear-material content a negative bias can be observed between the measured and calculated mass. Note however, that we do not have any information on the accuracy of the depletion calculations. Furthermore, the parameters used in the correlations for obtaining the concentration of nuclear material also need further investigation. The work on this subject is in progress.

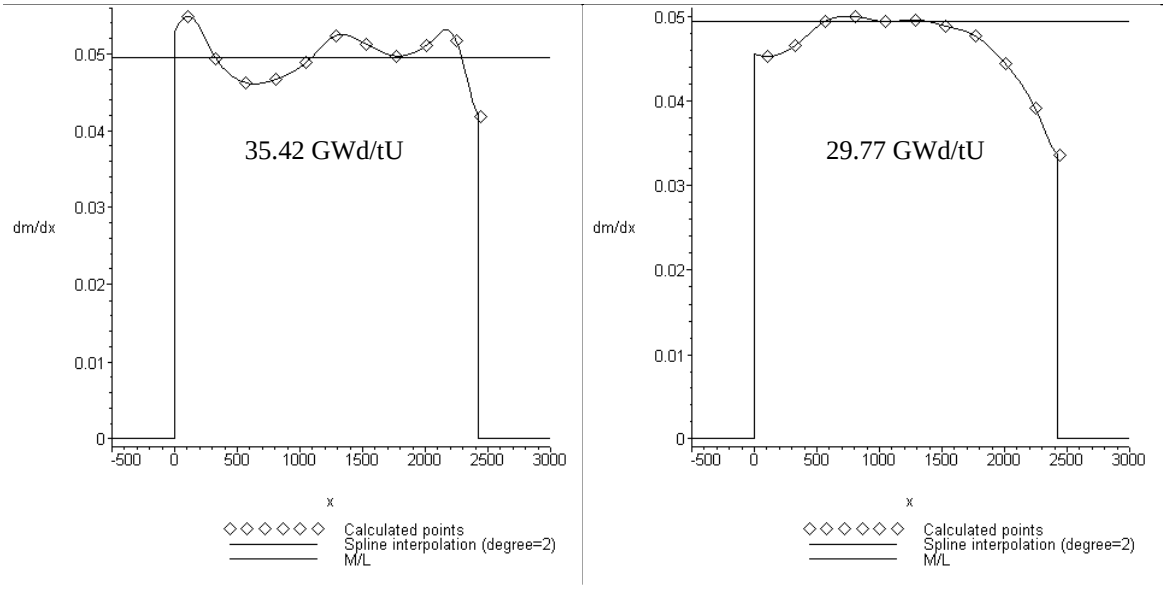



**Fig. 10.** The mass of spent fuel per unit length, as a function of the position along the assembly, for the two selected test assemblies from Table 1.

The type of measurements described above for determining the mass of nuclear material in a spent fuel assembly will also be used for calculating the mass of nuclear material in the containers containing the damaged nuclear fuel. In that case, however, the uncertainty of the calculated mass will also include an effect which could be called the mixed (or effective or apparent) burn-up effect, meaning that the burn-up of the mixture of damaged fuel is not well defined. In fact, it is neither proportional to the measured Cs activity ratio, nor is it correlated in a simple manner to the neutron count rate. Nevertheless, with certain limitations, accepting a somewhat higher experimental uncertainty, the Cs activity ratio still can be used for estimating an “apparent burn-up” which can then be used in the calculations. Results based on Monte Carlo simulations show that the error due to the

mixed-burn-up effect is a superposition of a systematic bias and of a random error. A more detailed study of this effect will be presented elsewhere.

### 5.3. The cooling time problem

For regular spent fuel assemblies the date of discharge from the reactor and the irradiation history are known. Consequently, a "cumulative cooling time" can be calculated and the values of the Cs activity ratio and neutron count rate can be corrected for the decay of $^{134}$Cs, $^{137}$Cs and $^{244}$Cm. For assemblies of unknown history, as well as for the damaged fuel placed into containers at Paks NPP, the "cumulative cooling time" could be estimated, in principle, from the $^{144}$Cs/$^{137}$Cs activity ratio. A more detailed study on the error of using such an estimate is in progress.

## 6. Verifying the contents of the nuclear-material-free containers

The containers which do not contain nuclear material but merely the heads and tails of the damaged assemblies will be verified for the absence of nuclear material. The containers will be scanned along their full length, using the device described above. The intensity of the gamma peaks, the neutron count rate and the gross gamma signal from the Si-diodes (not used during the described measurements in April 2006) will be indicators of the presence of nuclear material. If neither of the indicators is significantly larger than the smallest value detectable by the applied equipment, the container is to be declared "nuclear-material free", i.e. its nuclear-material content is smaller than the smallest detectable value. If either of the indicators is larger than the smallest detectable value, the container should be treated in the same way as the containers filled with nuclear material and its nuclear-material content should be determined by more detailed measurements.

The value of an indicator is to be regarded larger than the smallest detectable value if its deviation from the background is significantly larger than the (measurement time dependent) statistical error of the applied detector. For example, during the test measurements the neutron background in the spent fuel pond was 0,033±0,003 cps, which corresponds to about 10±1 counts within a 5 minute measurement. If one considers a signal to be significantly larger than the background if it is 10 times larger than the statistical error of the background, then in this particular case this implies that that the signal is significantly larger than the background if one records 10+10*1=20 counts within 5 minutes. This corresponds to (20 counts)/(300 sec)=0,07 cps.

Based on eq. (7) the lowest detectable mass of spent fuel can be estimated as

$$m_{uzemanyag}(\text{detectable}) \approx \frac{1}{\alpha}\frac{M}{L}\frac{N(\text{detectable})}{BU^{\beta}}\Delta x \qquad (11)$$

where $\Delta x$ is the length of the detected piece of spent fuel, assuming that the burn-up is nearly constant on this length. The corresponding nuclear-material masses can be calculated from eq. (4). Substituting $\Delta x$=1 cm (approximately the size of a pellet) and $BU$=17 GWd/tU then one obtains that the lowest detectable mass of spent fuel is 0.001 kg, that is, a single pellet is detectable if its burn-up is around average. However, if the same mass is smeared along the entire container ($\Delta x \approx$2420 cm), it cannot be detected. In that case the lowest detectable mass of spent fuel would be about 2.4 kg (corresponding to about 0.05 kg of $^{235}$U, 0.01 kg of Pu and 2.3 kg of total U).

## 7. Conclusion

Building the device described in this paper and performing the corresponding test measurements were initiated by the need to determine the nuclear-material content of the damaged spent fuel at Paks NPP. The purpose of the described measurements was to test and calibrate the device, and also to assess the performance of the method planned to be used for determining the nuclear-material content of the damaged spent fuel. It has been established that the difference between the measured mass of spent fuel and the mass known from the initial inventory of uranium is les than 10 %. The bias between the mass of nuclear material ($^{235}$U, total U and total Pu) is somewhat larger than this (up to ~ 20 %), which is probably due to the inaccurateness of the correlations used for the concentration of the nuclear material.

In this paper we only examined those sources of uncertainty, which are always there, i.e. even in the case of regular, undamaged spent fuel assemblies. In the case of the damaged fuel, being a mixture of pieces of spent fuel of different properties, a much larger source of error will be the uncertainty associated with determining the apparent burn-up of the mixture. The work on this subject is still in progress.

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# Session 22

# SAC Assurance

# French Domestic Safeguards inspections with regard to quality management system of the operators, in the field of nuclear material control and accountancy

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***Abstract:***

*The French Code of Defence states that the operators must follow-up the nuclear materials in their facilities and have to implement the technical conditions for nuclear material control and accountancy (NMC&A) under a quality management system which complies with the relevant international standards. This quality management system shall concern particularly: "receipts and shipments of nuclear materials, recognition of nuclear materials through checking and testing with a special attention on the related measurements, physical inventory taking of nuclear materials". These provisions lead French Domestic Safeguards to include operator's quality management system in the NMC&A verification program. The French safeguards inspections focuse particularly on movements and transformations of nuclear material, measuring and testing equipment, accountancy as well as identification and traceability, audits, nonconformities and defects management in the scope of NMC&A. This control is based on procedures contents and application, but above all on the associated document in proof. Such inspections are a good tool to assess the efficiency of interfaces between the nuclear material management, the operator accounting system and the national accounting system. This paper describes the verifications that are made by formal inspectors from French Domestic Safeguards and the lessons learnt. As a matter of fact, such controls provide wider information on the management of nuclear material in facilities over time and not only an overview of the inspection setting.*

**Keywords:** inspection; nuclear material control and accountancy (NMC&A); quality management system

## 1. Introduction

In France, the protection and control of nuclear materials is based on the Code of defence which has been published in December 2004. Concerning nuclear material control and accountancy (NMC&A), the technical conditions are set by the order of March 16$^{th}$ 2004. In the first article it is written that the operator shall implement a quality management system which complies with the relevant international standards, to day it means ISO 9001. In the field of material control and accountancy the operator shall implement a quality management system concerning particularly:

- receipts and shipments of nuclear materials,
- checking and testing with a special attention on the related measurements,

- physical inventory taking of nuclear materials.

The aim of this paper is to present the verifications made by the inspectors of the French National Safeguards to ensure that these measures are correctly applied.

## 2. Inspections with regard to quality management

### 2.1 Inspection's team and scope

The team of French National Safeguards inspectors (most of them coming from IRSN which acts as the technical support body of the Authority), includes generalists inspectors and specialized inspectors in measurements or accountancy. Furthermore some inspectors are trained to perform internal audits in IRSN and it is an interesting skill for inspections with regard to quality management.

In France the licensing is based on an authorization file, which described all the organization and provisions implemented by the operator to protect and control the NM. The preparation of the inspection is made with the help of the authorization file of the operator. All the others documents such as procedures or operating instructions can be seen during the inspection. The reference documents of the inspection is mainly the order of March 16$^{th}$ 2004.

The scope of the inspections is:

- general organization and responsibilities,
- review of the reference state of the facility,
- identification and traceability,
- control of nuclear material,
- accountancy,
- recording and storage,
- interfaces (mainly between control and accountancy),
- control of measuring and test equipment,
- internal verification and audit,
- measure, analysis and improvement.

All these items concerning the nuclear material management are reviewed taking into account the requirements of the ISO standard. The French National Safeguards control that quality management system and the rules implemented in organization and management which contribute to the quality of nuclear materials control and accountancy are established, implemented and maintained in order to improve the efficiency of this system. The areas covered are, at least, those mentioned in the order of March 16$^{th}$ 2004 (see Section 1) which, in practice, means that all activities related to nuclear materials control and accountancy, as well as their verification, shall be included in the quality system.

Following those inspections, the operators are required to inform the regulatory body (the High Civil Servant for Defence and Security at the Ministry for Industry) of the progress of these actions.

### 2.2 General organization and responsibilities

The national safeguards inspectors control that the specific organizational system set up for nuclear material control and accountancy is defined and described in the specific documents. In particular, the

main participants shall be expressly designated and their tasks shall be set out in the same way. The main participants are:

- the facility director, authorized to hold nuclear materials. In the event of a single company being present on several sites (i.e. COGEMA, EDF, CEA), the facility director is not the authorization holder (usually, it is the company manager) but the "appointed representative for the facility" of the company,
- the person appointed as the custodian of the nuclear materials by the facility director. This person is subject to legal obligations under the Code of defence (informing the authorities if any nuclear materials are assumed to be missing),
- staff members responsible for the control of nuclear materials,
- staff members responsible for nuclear materials accounting.

All these persons shall be qualified, having received suitable basic training, advanced training and/or have recognized experience. The national safeguards inspectors may check relevant documents related to training received.

### 2.3 Review of the reference state of the facility

In this part of the inspection the reference state of the facility should be reviewed; it consists in the documents submitted to the regulatory body for the licensing process which are mainly:

- the initial licensing and control file associated to the initial request for license,
- all the modifications to this initial file (change of authorized quantities of nuclear material, new activities…),
- provisions adopted for NMC&A,
- protocols established with others licensees for the shipments and receipts of nuclear material,
- documents designating the license holder and his appointed representative for the facility.

The role of the national safeguards inspectors is to verify that all these documents are maintained up to date but not to analyse them. The analysis of documents is an other process.

### 2.4 Identification and traceability

A quality system applied to the control of nuclear materials implies that they shall be identified in a consistent way, throughout their life in the facility. An identification system shall include a single content reference as well as a record sheet indicating at least the identity, the quality and the quantity of the nuclear materials. The sheet shall be affixed to the contents or in their immediate vicinity in the event of practical difficulties (glove boxes, shielded cells, size of contents) or for confidentiality requirements.

All data related to the nuclear materials records in the facility shall be maintained, in the form of documents. In these documents, what is concerned, who and when shall be specified. Regarding the sensibility of the NM, those documents are managed under specific confidentiality rules. It is particularly important that traceability be exhaustive, that is to say that all the interim storage, transformation and reprocessing phases be taken into account, from delivery to shipment of the nuclear materials in the facility. Likewise, it is essential that all the materials held be rigorously

controlled, including discards and waste, and not only those which constitute the main throughput of the facility.

It should be noticed that the identification and traceability of nuclear materials is rarely based directly on the materials themselves but more often on items in which they are incorporated (rods, fuel subassemblies and various set-ups) or on the different contents, in particular in the case of gases, liquids, powders, or waste. All modifications to the processing of nuclear materials shall therefore be carefully controlled. The national safeguards inspectors control that these requirements have been properly met.

## 2.5 Control of nuclear materials

All the movements and transformations of nuclear materials have to be accompanied by documents at each workstation in the facility, validated and transmitted to the centralized control system. The French Safeguards Inspectors shall control the corresponding procedures and operating instructions, their application and the global traceability. The licensee has the legal obligation to know at all times the quantities and qualities, the locations, uses, movements and transformations of nuclear materials in his facility; the effectiveness of the implemented system to attain this objective shall be demonstrated. The licensee shall also describe in procedures the check and testing of nuclear materials at delivery and, in particular the measurements related. These operations take place in two stages:

- during the 24 hours following the arrival of the nuclear materials the first level of control with fast non destructives assays (item counting, labels checking..)
- later on (but within the year following the arrival of the materials) the second level of control that determines accurately the quantities of nuclear materials is realized. When it is not possible, audits shall be performed by the licensee in the shipper organization. In this case, associated audit reports can be controlled by the inspectors. Generally speaking, these reports contain a lot of information and highlight any dysfunction that may have occurred between shipper and receiver.

All the discrepancies shall be treated in coordination with the shipper and transmitted to the national accountancy system.

Furthermore, the safeguards inspectors control with a particular attention the drawing up materials balances (material unaccounted for) and the calculation of their range of confidence as well.
Within the frame of control of NM, the licensee has the regulatory obligation to ensure that the assumed empty containers are really empty (vacuity procedure). The inspectors should control the associated documentary evidences.

## 2.6 Accountancy of nuclear materials

The implemented system of accountancy of the licensee shall maintain up to date the documents governing nuclear materials accountancy (such as the book inventory of the plant, the bookkeeping record of the movements) and allow the transmission in time of validated accounting declarations to the National Accountancy system (kept by the Institute for Radiation Protection and Nuclear Safety). The national safeguards inspectors make sure that these documents do exist and are applied adequately.

All the operating instructions related to « specific events » are examined such as inventory differences, material unaccounted for, disposal of N.M, accidental loss, shipper-receiver discrepancy, etc.

### 2.7 Recording and storage

A particular attention should be devoted to the verification of recordings.

The document storage rules shall also be described. The French regulation for NMC&A details the prescriptions for storage: most of the documents shall be retained in the facilities for 5 years after the nuclear materials concerned have leaved the site (for accountancy documents and their technical justification documents), some other shall be retained 5 years after a certified vacuity of the facility (detailed inventory report and their technical justification documents, bookkeeping records and documents giving evidence of the results of the associated comparisons).

### 2.8 Interfaces

An important item that inspectors could check is interfaces and descriptions of them. It includes particularly interfaces between physical control and accountancy, but also between local accounting system and National Accountancy system, as well as shipper and receiver of nuclear materials (cf.§2.5 and 2.6).

### 2.9 Control of measuring and test equipment

In order to control the NM, measuring devices are implemented, in relations with their form. The equipment concerned are those which are associated with the control of nuclear materials (weighing machines, on line measuring devices, analysis devices, computerized data processing systems, etc.). The objective of this control is to identify and put aside equipment which neither comply with specified requirements, nor with the associated methods.

During their inspections, the national safeguards inspectors can control, totally or partially, the following points (non-exhaustive list):

- that measuring systems and methods are described,
- that a measuring program exists, detailing by type of measurement, the frequency and the accuracy required,
- how consistent sampling is,
- the frequency of calibration and the availability of corresponding reports,
- calibration certificates of the standards used,
- the protection of test equipment against possible adjustments which would invalidate calibration setting.

### 2.10 Internal verifications and audits

Internal verifications are performed by the operator himself in order to control that provisions related in the different NMC&A documents are effective and correctly managed. These verifications consist of (non-exhaustive list):

- periodical physical inventories of nuclear materials, by sampling from inventory lists, counting, performing measurements and checking the presence and location of items, with regards to the control files. These periodical inventories are different from annual inventory required by regulation for each licensee, at least once a year,
- randomly verifications of the effective implementation of regulations,
- control of coherence between the book inventories in the local accounting records and the physical inventory as well as book inventories in the local accounting records and those in the centralized accounting system.

Besides, each licensee of nuclear materials shall be subjected to internal quality audits, organized by the management. The aim is to check the efficiency of the quality system implemented for nuclear material control and accounting.

During their inspections, the national safeguards inspectors generally check thoroughly the list of planned audits in the scope of NMC&A as well as the audit reports and corrective actions decided upon following these audits.

### 2.11 Measure, analysis and improvement

The approach for dealing with nonconformities and defects shall include research into their causes and shall call for corrective actions if necessary. The nonconformity and defect reports as well as the corrective action lists drawn up shall be maintained and may be assessed by national safeguards inspectors.

However, it’s worth noticing that corrective actions are requested, if needed, following inspections by national safeguards inspectors. These are systematically controlled by the national safeguards inspectors.

In order to evaluate the opportunity of improvement for the processes concerned, management reviews (formal evaluation, by the management, of the status and adequacy of the quality system in relation to quality policy) shall be organized. Inspectors could check that quality management reviews are performed at regular intervals.

In fact, the improvement of quality management system in the scope of NMC&A is directly or indirectly linked to the experience feedback ie lessons learnt from problems that have already occurred in the facility or in other facilities (from the same company for example).

## 3 Conclusion

The wants in traceability, responsibilities, recording, archiving, to attain the objectives as well as the continual improvement introduced in the quality system reach the expected aim so as to get an efficient NMC&A system. Their introduction in the NMC&A regulation provides to the operator a framework to control NM (a framework that the operator uses in order to manage the quality management system). It allows the inspectors to point out the recorded history of the items, the process at work and also to determinate any responsibility in case of dysfunction. So, this is all to keep

in mind, that the aim of the inspection mission goes further than quality audits and allow to control that the dispositions implemented by the operator guarantee the protection and control of NM.

The lessons learnt from documentary evidences assessment, from internal verifications and audits as well as measures, analysis and improvements are particularly significant. Indeed, such controls provide wider information on the management of nuclear material in the facility over a chosen period and not only a particular overview of the inspection setting.

The organization set up for NMC&A by the operators provides more confidence for inspectors.

The implementation of an appropriate quality management system impose much more thoughtful and efforts from the operators. However, for French National Safeguards inspectors the implementation of quality management in the regulatory requirements for NMC&A is indeed associated with a regulation more closely followed, a better management of nuclear materials (near-real-time accountancy, identification of materials not being used, etc.), a reduction in dysfunctions at the interfaces of activities and an improved experience feedback.

# Conclusions of the ESARDA focus group on NMAC: Application of audit techniques in international safeguards

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## Abstract

*The European Commission has been in the process of reviewing the implementation of nuclear safeguards under chapter VII of the Euratom Treaty and aims to place greater emphasis on establishing, by audit, the adequacy of the operator's own nuclear material control and accounting systems. There are concerns on how this changes the activities of safeguards verification and how this may interact within the overall quality management systems of nuclear facilities.*

*Therefore an ESARDA working group was formed to clarify a range of issues and to produce a guideline reference against which an objective assessment of an NMCA system could be made. This paper summarises the work of the focus group based on three key deliverables: an advisory document, an NMCA guideline document and a guideline for conducting safeguards audits at nuclear facilities. In considering the issues the working group sought synergies not just with the quality assurance audit environment but also with the risk based environment of financial audits.*

## Introduction.

An ancient Greek quote sums up the modus operendi of the Euratom safeguards verification regime until recently "There is only one safeguard known generally to the wise – suspicion".

Universal non-discriminatory suspicion has led to significant complexity in European safeguards approaches. The safeguards technological "silver bullet" in itself can not be the only response. A modern approach needs to give some recognition and accommodation of, the operator and states systems and of the confidence building that has taken place over the years.

A high level expert review of Euratom safeguards [1] advised that in the integrated Europe of today there was "no point in keeping alive a system that implies a high degree of suspicion" towards EU states. The Commission subsequently declared a revised strategic direction for Euratom safeguards [2] in which it indicated that it will "assess the fitness of the operator's systems for the purpose of detecting with high assurance in a timely fashion any losses or apparent losses". The implementation of this strategy has been discussed with Member States and resulted in agreement in February 2007 of the Commission document, "Implementing Euratom Treaty safeguards" [3]. The Commission expects to be able to assess the performance and risks of a nuclear installation in order to decide whether to include audit of the installation in its annual programme of inspection activities.

The traditional safeguards paradigm which centred on comparing accounts with the real presence of material needs to be integrated into a more flexible approach. Integrated safeguards, international fuel cycle centres and international recognition of state level approaches are all elements of this evolution. This evolution requires international input from operators and national authorities and the assurance needs of the international community.

## Strong positive safeguards assurances.

The Safeguards' challenges of the future will depend on the type and scale of nuclear power that exists and the prevailing public and political opinion towards it. The current position on global warming, security of energy supply and the increased costs of energy make a renaissance of nuclear power in Europe in the next decade more likely. Strong, independent safeguards assurances are an important enabler in gaining public acceptance and benefit both industry and the wider community. The nuclear industry is aware of the importance of allaying all nuclear risk concerns as early as possible in the lengthy public consultations that accompany the construction and operation of major nuclear build projects. The high level expert group comment on assurances was that whilst facilities are responsible and accountable

for nuclear material management, Euratom safeguards should maintain a “close watch over this safekeeping” and “this will keep nuclear materials out of the headlines”.

## State the problem and state the solution.

What is the problem for which audit is the solution? From above, we can see that audit is part of the solution for gaining assurance of good operator performance and capability. We can also see that in the nuclear build context the additional assurance could be very valuable.

However, in the “old” member states of Europe, Euratom safeguards verification, in all types of fuel cycle facility, is already mature and effective, and the inspectorate has an extensive knowledge of past and current performance and on the operators’ systems. For such operators, the perceived benefit of an increased assurance statement is not proportional to the possible effort involved in audits. In addition, the expansion of the European Union from 10 Member States to 27 has brought with it nuclear installations and operators for which the Commission has no track record. Audit is a tool that can be used to assess system capability whilst experience via independent verification is being built up.

Finally, something like 60% of the world's safeguards resources are deployed in Europe. Audit offers potential for a better targeted use of inspection resources and a move towards risk based and more holistic approaches. Operators continue to be willing and able to work with the Commission to help identify new, more efficient ways of achieving the assurance of declared use, as they have done in the past. Their aim is to have a healthy and positive relationship between inspectors and inspected. Operators and authorities must all work to identify cheaper, smarter and more flexible safeguards systems that are less intrusive and mechanistic.

## NMAC audit focus group (NMACAF).

In 2005, the European Commission sought advice from ESARDA on the role of NMAC audit in the safeguards approach and consequently a focus working group was formed. The group had 6 meetings, the first in March 2006 and the last in March 2007. The overall objective of the working group was to provide the safeguards community with expert advice on NMAC systems’ good practice and the use of audit methodology applied to such systems in support of safeguards verification.

The working group has delivered three outputs. Firstly, a guideline reference for good practice NMAC [4], against which to audit. Secondly an advisory document on the concepts and general implementation of audit [5], and thirdly a guideline on the conduct of safeguards audits [6].

The group did have diverse views and even a definition of safeguards audit was a matter of some debate. The advisory report suggestions should therefore not be regarded as unanimous on all points.

We should also ask the same problem/solution question of the ESARDA focus group. The working group was presented with a variety of issues and concepts needing to be clarified. Concerns about audit included:-

- the value of audit;
- information security;
- escalation of the safeguards burden;
- added layers of bureaucracy;
- challenging management effectiveness;
- impact on operations, resources, costs;
- relevance in the joint team environment.

## The NMACAF advisory report.

The advisory report covers:-

- Systems audit in a Safeguards context and as a safeguards tool;
- Justification for audit, the legal framework and impact on PSP, BTC & special reports;
- Impact on operator and key stakeholders and interaction with National authority;
- Other relevant quality/audit frameworks and taking them & accreditations into account;
- Guidance on NMAC good practice and on the frequency and conduct of audits and the measures appropriate for the scale of nuclear operations;
- Success criteria, assessment tools, performance metrics, utilisation of results of audits and potential for benchmarking;
- Competency, language, support and training requirements;
- Impact on inspection objectives, detection capability and physical verification. Impact on the IAEA and on Integrated Safeguards (IS).

As in the case of the IAEA 93+2 programme for strengthening safeguards there is a need to clarify the legal position of what can be done within the existing legal instruments. Where performance issues are protracted or systemic, then the Commission should offer the operator the option of allowing a full system audit thus taking an educational rather than punitive approach. Audit should be implemented with a code of practice (the guidelines) and in accordance with the principle of proportionality. Audits must be targeted and applied consistently, equitably and pragmatically and used sparingly. Installations with good performance under traditional verification and demonstrated compliance of their NMAC approach should be able to reject audits if the burden is disproportionate to the benefit.

In order to make progress on the deployment of audit it is first necessary to suitably define what the Commission means and intends by audit. Well formulated criteria for assessing confidence and risk, need to be in place and transparent to operators and member states. The Commission should set itself a time limit to gather such information and formulate its methodologies. This should include the tailoring to installation specifics such bulk handling .v. item handling, large .v. small installations, centralised .v. distributed organisations, manual .v. automated, old .v. new etc.

Audits in the conventional sense, are mostly a top down process concentrating on management and infrastructure issues. Assessment of NMAC compliance and performance could be approached as a bottom up process. It may be therefore, more appropriate to separate these approaches rather than tackle them simultaneously under the banner of Safeguards audit.

## Capitalising on state/operator systems.

The reality of the situations is that audits can consume some 3 times the Personnel Man Days of Inspection (PDIs) as normal inspections, unless some efficiency measures are deployed. In real terms this means taking a more flexible and less independent approach and working with what an operator or state already has in place. This includes state level quality assurance programmes and NMAC standards/guidelines, internal and external audits of installations NMAC and certification and accreditation awards. State authorities should take part in the Euratom safeguards audits and follow up. Member states should progress the follow up and could share the results with the Commission.

There is scope for recognition of these factors and their reflection in a reduced level and frequency of verification. The better and more transparent the system, the higher the level of confidence and hence, increased potential to lower the level of routine, mechanistic nuclear material verification. The emphasis should no longer be exclusively on independent systems but should include accreditation, authentication and random testing. The impact of audit on the Euratom safeguards regime is hard to quantify as it will depend on factors such transparency, openness and unpredictability. Audit is however, not a substitute for physical verification. This must be maintained at a level which provides credible safeguards assurance of non diversion. There is a need to formalise the process for marrying verification and audit findings, and deriving a confidence level which could influence inspection activities towards a (pre-) defined minimum credible level. Until that time the Commission should not embark on inspection reductions based on audit findings. Under a quality approach, any timeliness criteria could expand and contract in line with confidence levels and uncertainties/anomaly detection.

## Quality assurance.

Audit can be an effective method for driving continuous improvement. Commission auditors should therefore approach audits in a spirit of collaboration and employ a different cultural approach to that used for physical verifications. Substantive reviews should be conducted as part of a collaborative framework of peer review of the operator's system and operators should be encouraged to voluntarily go beyond the confines of the regulation in order to show active quality management. Such peer reviews are best conducted in a confidential voluntary framework and should have clearly defined expectations and performance objectives. The Commission should however be transparent about its audit activities and the effect on PDIs (whilst retaining operator anonymity).

The biggest added value from most audits comes from obtaining a clean bill of health. The Commission should give a positive endorsement in their audit findings when appropriate.

## What to audit against.

The technical component of audit is to meet criteria and to have a system of evidence to substantiate the quality of products and processes. Whilst the Euratom Regulation embodies many compliance statements it simply calls for a description of the installation's nuclear material accountancy and control system "describe item and/or mass accountancy system, including assay methods used and assessed accuracies, supplying specimen blank forms used in all accountancy and control procedures. Period during which such records must be retained should be stated". This is not an adequate basis for assessing the quality and performance of the NMAC system.

In seeking quality assurance, the working group looked to the international standards organisation and in particular to ISO 9001/2000 which is written so generally that any process can be poured into its mould. This level of abstraction is still little better than that in the regulation. If this was considered appropriate then there would be no need for any other standards or guidelines for safety, environment, finance etc. The reality is that the basic principles and practices which make up NMAC must be defined before any assessment of system capability can be made In any specific facility situation there can be a variety of different approaches that are equivalent as far as good practice is concerned and that meet the regulatory requirements. For this reason imposition of a prescriptive technical approach to NMAC (other than fitness for meeting regulatory requirements) is inappropriate. The working group however did consider it appropriate to produce guidelines on good practice.

## NMAC good practice.

The working group produced NMAC good practice guidelines, which are now available for view on the ESARDA website and are considered a live document for utilisation and developing further. In practice, the guidelines need to be applied to the specific features of each facility's approach to NMAC. This can be done by identifying the processes of the specific facility approach and the guidelines then indicate the types of performance objectives that can be of relevance to audit of those processes.

Since it is a matter of choice for individual facilities whether they wish to manage NMAC through a formal quality management approach or not, the guidelines provide criteria and approaches for good practice that can be taken into account whether or not the facility is following a formalised quality management scheme.

The guidelines express criteria and identify technical issues. They take account of the need for NMAC solutions to be adapted to the technological reality the objective safeguard risk and the burden to the operator. By drawing on the experience of facilities, they offer the possibility to meet the criteria of the regulations and PSPs in a flexible way (this is important for example in old plants).

The guidelines distinguish between "item facilities" where measurement is not involved and processing facilities where material changes form and material balance must be assessed taking account of measurement uncertainties. The guidelines emphasise the fact that NM control can be achieved in a simple way for item facilities and outlines the requirements for both types of situation.

For situations in which the effectiveness of the facility control system is dependent on the effectiveness of complementary measures, e.g. monitoring (C/S) serving other purposes, the guidelines elaborate the logical conditions that such monitoring should satisfy in order to provide control assurance.

The guideline can be used both as a basis for orienting audit as well as being useful for focusing self-improvement strategies. If audit provides assurance that the NMAC and reporting process is run in compliance with agreed procedures, the confidence will be high that material control and safeguards reporting will meet the regulatory requirement.

The guidelines are formulated in such a way that, if some of the operators NMAC processes are certified to a management system standard or to a technical competence standard, audits could make use of the results of such certification. The guidelines can assist in identifying the criteria for recognition of results of certification audits.

## The content of the NMAC guidelines.

The guidelines describe objectives, issues and good practice for the following types of NMAC process.

*Measurement*: This includes measurement of process materials for accountancy as well as the activities of the facility measurement control program. The guidelines identify issues covering the measurement control program, estimation of measurement uncertainty parameters and the maintenance of instrumentation. It also includes archiving of measurement data, measurement control results and records of maintenance and repair activities so that traceability requirements are met.

*Tracking and Control of Material*: The guidelines describe objectives and good practice for nuclear material tracking and inventory control. It covers the creation of transfer and stock data that provide information needed in meeting regulatory requirements. This includes objectives for record keeping in material follow-up and in particular the use of material control zones to ensure a reliable running inventory. The guidelines elaborate the different approaches to material control that distinguish item control zones from mass accountancy zones. It also discusses quality issues related to inventory verification by the operator.

*Data Storage and Processing*: This includes all data storage and processing activities linked to the provision of accounting reports, or to inventory control or supporting physical inventory verification (LII). While discussing data processing to support obligations under the regulation, the guideline discusses traceability of information and in particular traceability in regard to errors and correction of errors. It also elaborates the main considerations regarding effective management of the DP function such as software control, data control, DP access control and monitoring of DP activities.

*Accountancy Balance Procedures*: The guideline describes objectives and good practice for all procedures involved in checking whether all Nuclear Material is accounted for. This identifies issues concerned with receipts, shipments, physical inventory taking (PIT) and the method for assessment of the material balance value. In discussing PIT, the guidelines emphasise the need to limit the work and radiation costs involved while nevertheless providing a reliable result. They identify and discuss QA and QC issues for inventory control when running inventory is to be used as a basis for PIT. This covers quality in the recording of transfers and storage as well as assurance the completeness of records and the continuous validity of the material description. This discussion is related to the discussion of a follow-up approach that is based as much as possible on item control.

*Management Supervision of NMAC Processes*: The guideline describes how measurement, material tracking and data processing are themselves normally the object of explicit procedures of quality assurance and control carried out by technical staff. Any of these three basic activities or their related QA and QC processes can be assessed for the quality of their performance. Any of these processes could be assessed in terms of the usual quality issues involving methods, documentation and training, failure risks, recognition and treatment of failures, record keeping and responsibilities relating to all of these. The guideline seeks to ensure that these concerns are adequately managed by the facility itself through the effectiveness of management supervision applied to both basic activities and their control processes. The important functions in supervision are the capacity to recognize whether unacceptable performance deficiencies exist, to identify an appropriate response and to implement change. This recognition and response capacity will usually be incorporated in the role of technical quality staff and operational management. This supervision and response capacity may or may not be part of a formal quality management system.

## Harmonisation.

Given an increasing importance of quality matters in the EC safeguard approaches, it is important to avoid confusion over terminology. Consistent and high quality use of NMAC and audit terminology should be supported by a terminology database.

Each installation should also be encouraged to harmonise NMAC system processes across its operating and business units so as to reduce complexity, enhance technical integrity, increases transparency and better enable efficient audit.

It would be good practice for operators to have a concise statement of their own NMAC implementation model including how and where it overlaps with installation quality management processes. Since good practice can only be ‘voluntary’, then the Commission should accept and record the operator’s rational as to why a particular good practice should not apply in their particular circumstance.

It needs to be explored whether the NMAC guidelines can be further laid out in a pattern tailored to specific plant types (as in the BTC). In anticipation of new reactor build the NMAC guidelines have an annex for NPPs.

## "Keeping the score".

The main tool for assessing NMAC performance will continue to be the material balance; the completeness and correctness of flow and inventory data and the installation's measurement capability (bulk handling) and tracking capability (item handling). Any objective assessment of quality must be based on evidence, "keeping the score". There should be metrics for performance and in the first instance the burden of metrics should lie with the Commission. Where operators offer access to an installation's own performance monitoring system then those indicators should be taken into consideration and randomly checked for reliability and authenticity.

Good practice in the sense of target values may be defined for measurement/analytical methods and possibly for other aspects of specific NMAC approaches in selected sections of individual plants. To do this, the Commission needs to establish with each operator, equitable target benchmark values for NMAC accuracy and timeliness and for detection capability appropriate to the installation and material type and should benchmark an installation across its own operations and performance history.

## Out of scope?

Factors at play in quality and performance assurance of NMAC and safeguards systems in nuclear facilities often appear at first to be purely technical in nature but are in practice a mix of attitudinal, resource, radiological and technical factors. NMAC audit is concerned only with how the activities provide the NMAC results. NMAC activities can be assessed in terms of regulatory conformity and fitness for use.

Quality Management is a common business process and measures the pulse of how successfully a business is run. Assessment of effective quality management include consideration of factors such as governance, leadership, empowerment, communication, competencies, learning, continuous improvement, commitment to a common sense of purpose, integrity and resilience. Audit at this level goes beyond that necessary for detection of diversion or compliance with international agreements.

The evaluation of information relative to the hypothesis that "there are no quality issues in NMAC" requires that all the information be placed in a context or structure that makes it possible for the auditor to associate indicators with activities and to recognize ambiguities or inconsistencies where they exist. In this regard wider breadth and depth audits provide increased transparency. By being clearly and precisely scoped however, they can provide confidence that sensitive technology details and commercial information will not be compromised.

A fundamental paradigm shift for Euratom safeguards only occurs where the scope of audit extends deep into the management system or where audit significantly supplants physical inspection. Independent verification by safeguards inspectors is unique and is not found in typical ISO (9001) style audits. Audit is not a substitute for such activity. Audit can however help define risks, which in turn can be used to determine the level of inspection required and consequent efficiencies. The better framework for audit is where the Commission would stimulate the broader need for achievement and self control

## Conclusions.

Auditing should have a place within the framework of the Commission's inspection role under the Treaty. It will provide more insight into an operator's systems and a more holistic way of assessing the NMAC and safeguards procedural framework and quality control mechanisms. Audit methodology can also identify the information appropriate for facilitating the evolution of safeguards approaches;

How far and how fast the Commission goes down the audit path will depend on

- the results from audit field trials and on the technical and cost effectiveness of audit.
- the determination of Commission and member states to develop the audit regime.
- the acceptance of audit measures by the IAEA, the member states and the operators.

In 2003, Mr Burrows, as the then chairman of ESARDA, presented a paper on making a difference in safeguards [ref 7] in which he said

that ESARDA had access to all aspects of relevant “areas of competence'' and acted as a catalyst for improvements in safeguards:-

- formulating methodologies and technical projects for safeguards development;
- developing international benchmarks and standardised approaches;
- supporting activities which improve quality and performance;
- advocating evidence-based policies;
- raising public appreciation, confidence and trust in safeguards;
- developing a professional identity for safeguards practitioners;
- fostering a safeguards culture of openness and transparency;

We believe this was the case in the NMACAF working group and hope that the future will judge the work of the group to have made a difference.

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# Session 23

# Environmental Sampling & nuclear forensics

# Determination of Long-lived Radionuclide Impurities in Uranium-oxide

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***Abstract:***
*A novel and simple destructive sample preparation method has been developed for the simultaneous measurement of $^{230}$Th and Pu-isotopes in uranium-oxide matrix by isotope dilution inductively coupled plasma sector-field mass spectrometry (ICP-SFMS). The procedure allows the determination of the production date (age) of the nuclear material and also enables the detection of the presence of irradiated or reprocessed fuel in the sample. The method developed was compared with the quasi non-destructive laser ablation ICP-SFMS technique in order to measure directly the Th and Pu-content. The advantages and disadvantages as well as some possible applications in the field of nuclear forensic science are presented.*

**Keywords:** illicit; nuclear; age determination; laser ablation; ICP-SFMS

## 1. Introduction

Identification and characterization of found or confiscated nuclear materials, such as uranium-oxide or plutonium-oxide matrices, form integral part of combating nuclear smuggling and illicit trafficking [1,2]. Applying advanced analytical techniques the radiological hazard, intended use and possible origin of the nuclear material can be assessed. Beside the commonly measured parameters, such as dimensions, U or Pu content, isotopic composition or trace impurities, also the production date ("age") of the nuclear material, i.e. the time elapsed since the last chemical separation of the daughter nuclides from the mother radionuclide (usually U or Pu) can be determined [3-5]. This unique possibility is based on exploiting the decay of the long-lived radionuclide content. The production date helps to reveal the origin of the nuclear material and to trace back the possible route of the sample to its source allowing the authorities to improve protection measures.

Nuclear samples (e.g. fuel pellets, intermediate products or ore) contain various types of trace-level impurities (Table 1).

| Impurity | Origin | Measurement methods | Information | References |
|---|---|---|---|---|
| Stable elements | Starting material, chemicals added, production method | ICP-OES, ICP-MS, GD-MS | Production location, origin, production method | [1,2,6] |
| Decay products | Decay of base material | AS, GS, TIMS, ICP-MS, SIMS | Production date | [1,2,5,7,8] |
| Activation products | Irradiation | AS, SIMS, ICP-MS | Reprocessing, burn-up | [7,9] |
| Fission products | Irradiation | GS | Reprocessing | [10] |

*ICP-OES: inductively coupled plasma optcal emission spectrometry; ICP-MS: inductively coupled plasma mass spectrometry; GD-MS: glow discharge mass spectrometry; AS: alpha spectrometry; GS: gamma spectrometry; TIMS: thermal ionization mass spectrometry; SIMS: secondary ionization mass spectrometry*

**Table 1:** Trace-level impurities in nuclear materials.

Stable elements, such as transition or rare-earth elements are introduced into the material mainly from the original starting material (e.g. ore), from the added chemicals used for the

production and also as a result of contamination during the production route. Decay products of the nuclear materials are continuously built-up after the last chemical separation in the material due to the decay of uranium or plutonium. Activation (e.g. $^{236}$U, $^{239}$Pu or $^{240}$Pu) and fission products (e.g. $^{137}$Cs) can also be present in the nuclear materials if it has been previously irradiated or mixed (e.g. after reprocessing) with irradiated uranium.

Analyses of trace-level impurities play an important role in order to deduce back the possible route of confiscated illicit nuclear material. These parameters give information on the date (age) and location of production as well as on the method used for the nuclear fuel production, thus determination of these parameters are of high importance in nuclear forensic investigations.

The aim of this study was to develop a simultaneous and simple destructive sample preparation method for the determination of $^{230}$Th (production date determination) and Pu-isotopes (information on reprocessing) from uranium-oxide material by isotope dilution inductively coupled plasma sector field mass spectrometry (ICP-SFMS). The method developed was compared with the direct, quasi non-destructive laser ablation ICP-SFMS technique. The advantages and limitations of both methodologies together with possible applications are presented.

## 2. Experimental

The experimental methods and conditions can be found in details elsewhere [8]. In short, for the destructive method the $^{230}$Th and Pu-content of the dissolved uranium-oxide was separated with extraction chromatography using TEVA$^{TM}$ resin. The pure Th and Pu-fractions were analysed by isotope dilution ICP-SFMS using $^{233}$U, $^{229}$Th and $^{242}$Pu isotopic tracers. The mass spectrometric analysis was carried out using a double-focusing magnetic sector inductively coupled plasma mass spectrometer equipped with a single electron multiplier (ELEMENT2, Thermo Electron Corp., Bremen, Germany). Measurements using liquid sample introduction were carried out in low resolution mode ($R$ = 300) with a low-flow micro-concentric nebulizer operated in a self-aspirating mode (flow rate was 100 μl min$^{-1}$) connected to a stable introduction system spray chamber (SIS, Elemental Scientific, Omaha, USA) in order to improve isotope ratio precision. The schematic diagram of the destructive sample separation for $^{230}$Th and Pu-isotopes determination is shown in Figure 1.

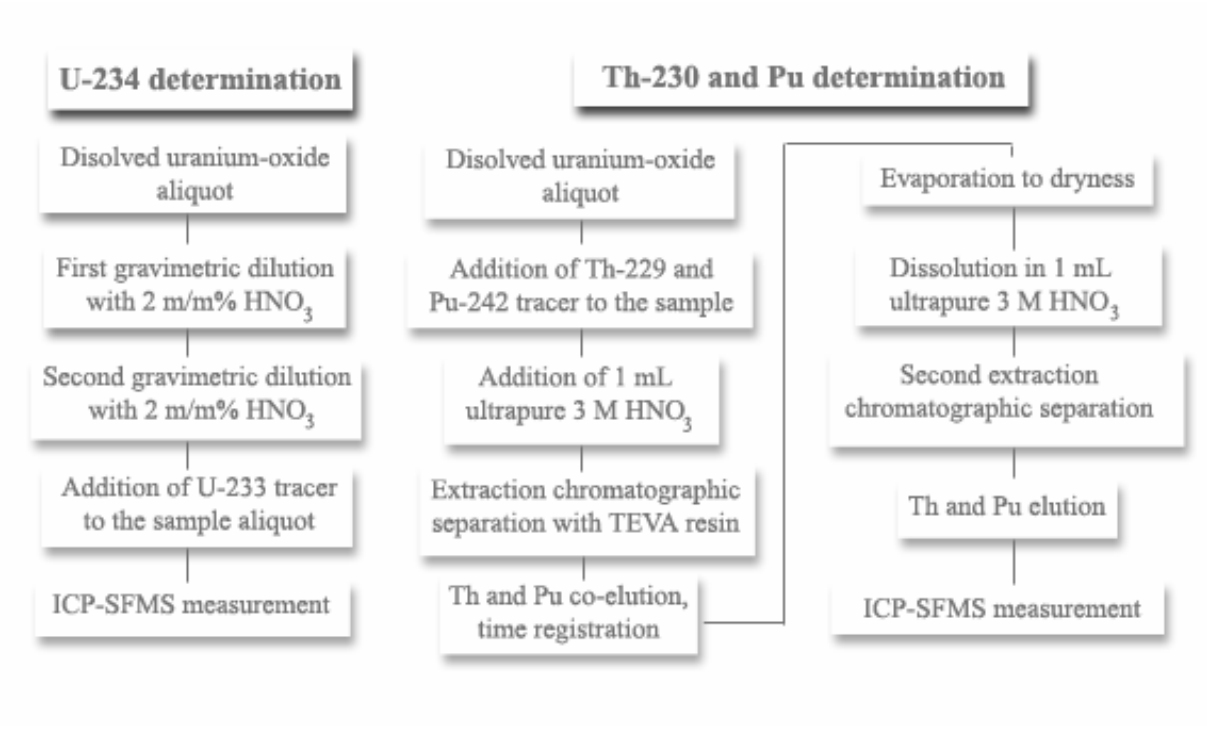


**Figure 1:** Schematic of sample preparation for destructive Th and Pu determination in uranium-oxide.

The direct laser ablation studies for the age determination (by the measurement of $^{230}Th/^{234}U$ ratio) and the measurement of the Pu-isotopes were carried out using an UP-213 laser ablation system (New Wave, Freemont, USA). The ablated material is transported by argon as a carrier gas into the plasma. Preliminary studies showed that medium resolution ($R$ = 4000) is necessary to achieve accurate age determination results by the measurement of $^{230}Th/^{234}U$ [8]. In case of plutonium determination by LA-ICP-SFMS similarly to the liquid introduction method interference from the $^{238}U$ signal is the main source of interference [11]. Furthermore, in order to reduce uranium load of the plasma and to avoid contamination of the instrument $^{238}U$ signal was kept lower than $10^9$ cps. Concentrations of isotopes of interest measured by liquid sample introduction were calculated as a function of $^{230}Th/^{229}Th$ and $^{234}U/^{233}U$ ratios according to the isotope dilution method. All raw data were corrected taking into account instrumental mass bias using linear correction. The overall uncertainty was calculated taking into account the uncertainty of the weight measurements, tracer concentrations, measured intensities and half-lives according to ISO/BIPM guide.

## 3. Analysis of confiscated uranium-oxide materials

### 3.1. Age determination of uranium-oxide

Three uranium-oxide pellets or materials confiscated in Hungary were analysed in this study. The samples have different $^{235}U$ enrichment: HU-DEP is a depleted uranium pellet ($^{235}U/^{238}U$ isotope ratio is 0.00258 ± 0.00004), HU-NAT is a natural uranium containing material, presumably a nuclear by-product ($^{235}U/^{238}U$ isotope ratio is 0.00713 ± 0.00014), while HU-LEU is a low-enriched uranium pellet ($^{235}U/^{238}U$ isotope ratio is 0.0255 ± 0013). For the laser ablation investigation, uranium-oxide standard (referred to as UOX-STD) was prepared from natural uranyl-acetate (Lachema Ltd., Czechoslovakia). The $^{230}Th/^{234}U$ atom ratio in the sample was measured with the destructive method. In order to validate the methods, highly-enriched uranium-oxide powder from a Round Robin interlaboratory exercise (RR-HEU) organized by the Nuclear Smuggling International Technical Working Group (ITWG) was used [12]. For the destructive analyses the known amounts of uranium-oxide materials were dissolved in 6 M ultrapure nitric acid while heating slightly in a water bath. For the laser ablation measurements the pellets were measured directly, while powder samples were pressed hydraulically into a disk-shaped pellet with a diameter of 5 mm.

The production date determination of the uranium-oxide material is based on the decay of the relatively long-lived $^{234}U$ ($T_{1/2}$ = 245250 ± 490 a) to $^{230}Th$ ($T_{1/2}$ = 75690 ± 230 a) and the disequilibrium between these two radionuclides. After the last chemical separation of $^{234}U$ during the preparation of the nuclear material, the concentration of $^{230}Th$ daughter nuclide is continuously increasing in the uranium-oxide material. The theoretical $^{230}Th$ amount formed by the decay can be calculated by use of the equations of the radioactive decays assuming that the initial concentration of the daughter nuclide is zero after the last chemical separation (i.e. the separation was complete) and utilizing the approximation that $^{234}U$ amount in the sample is constant over the investigated time scale. These assumptions are usually valid for nuclear fuel samples. If the atom ratio of $^{230}Th$ and $^{234}U$ is measured, the elapsed time and the production date can be calculated as follows:

$$t = \frac{1}{\lambda_{U-234} - \lambda_{Th-230}} \ln\left(1 - \frac{N_{Th-230}}{N_{U-234}} \cdot \frac{\lambda_{Th-230} - \lambda_{U-234}}{\lambda_{U-234}}\right) \quad (1)$$

where $N_{Th\text{-}230}/N_{U\text{-}234}$ is the atom ratio in the sample, $\lambda_{Th\text{-}230}$ and $\lambda_{U\text{-}234}$ are the decay constants of $^{230}Th$ and $^{234}U$, respectively, and $t$ is the elapsed time since the separation of the radionuclides.

The $N_{Th\text{-}230}/N_{U\text{-}234}$ atom ratio and also the production date are determined in the investigated samples by the destructive isotope dilution and quasi non-destructive laser ablation ICP-SFMS method. The $N_{Th\text{-}230}/N_{U\text{-}234}$ atom ratio can be calculated from the $^{230}Th$ and $^{234}U$ concentrations obtained by the destructive method (Table 2) [8].

| Sample | $^{235}U/^{238}U$ ratio | Measured $^{230}Th/^{234}U$ atom ratio | Calculated age (a) | Production date |
|---|---|---|---|---|
| HU-DEP | 0.00258 ± 0.00004 | $3.88 \cdot 10^{-5} \pm 2.8 \cdot 10^{-6}$ | 13.7 ± 1.0 | August, 1993 (± 12 months) |
| HU-NAT | 0.00713 ± 0.00014 | $5.01 \cdot 10^{-5} \pm 2.4 \cdot 10^{-6}$ | 17.73 ± 0.87 | July, 1989 (± 10 months) |
| HU-LEU | 0.0255 ± 0.0013 | $4.38 \cdot 10^{-5} \pm 2.8 \cdot 10^{-6}$ | 15.51 ± 0.98 | September, 1991 (± 12 months) |
| RR-HEU | 10.80 ± 0.17 | $7.89 \cdot 10^{-5} \pm 3.6 \cdot 10^{-6}$ | 27.9 ± 1.3 | June, 1979 (± 16 months) |

**Table 2:** Production date of the investigated samples measured by the destructive ICP-SFMS method.

Though in case of laser ablation measurements the intensity ratio of $^{230}Th$ and $^{234}U$ can be directly obtained from the mass spectrum, correction has to be applied in order to convert it to $^{230}Th/^{234}U$ atom ratio, which is necessary for age determination. The direct determination of $^{230}Th/^{234}U$ atom ratio from the mass spectrum using laser ablation is hindered by the slightly different ablation and ionization efficiencies of $^{230}Th$ and $^{234}U$, however, this effect is not as significant as for SIMS measurements. For the calculation of isotope ratios, relative sensitivity factor (RSF) approach is commonly applied, which takes into account the differences in the ionization and ablation efficiencies of the elements at fixed conditions (constant laser energy, crater diameter and repetition rate). The RSF value can be calculated as follows:

$$RSF = \frac{S_{Th-230}}{S_{U-234}} = \frac{I_{Th-230}}{I_{U-234}} \cdot \frac{N_{U-234}}{N_{Th-230}} \cdot \frac{M_{U-234}}{M_{Th-230}} \qquad (2)$$

where $S_{Th-230}$ and $S_{U-234}$ are the sensitivities of $^{230}Th$ and $^{234}U$, $I_{Th-230}$ and $I_{U-234}$ are the measured intensities of $^{230}Th$ and $^{234}U$, $N_{U-234}/N_{Th-230}$ is the atom ratio in the sample, $M_{Th-230}$ and $M_{U-234}$ are the molar weights of $^{230}Th$ and $^{234}U$, and $c_{Th-230}$ and $c_{U-234}$ are the concentrations of $^{230}Th$ and $^{234}U$ in the samples, respectively. If the RSF value is experimentally determined using a uranium-oxide standard with known $^{230}Th$ and $^{234}U$ content, the $N_{Th-230}/N_{U-234}$ atom ratio in the investigated sample can be calculated (Table 3) [8].

| Sample | Calculated age (a) | Production date |
|---|---|---|
| HU-DEP | > 2.3 | Earlier, than November, 2004 |
| HU-LEU | 16.1 ± 3.0 | February, 1991 (± 3 years) |
| RR-HEU | 30.4 ± 3.8 | October, 1976 (± 3.8 years) |

**Table 3:** Production date of the investigated samples measured by the laser ablation ICP-SFMS method.

The age results of uranium-oxides obtained agree with those measured by the destructive isotope dilution ICP-SFMS method and also with the previously reported values for the Round Robin sample (February to July, 1979) [12].

## 3.2. Determination of plutonium in uranium-oxide materials

The Pu-isotopes in the investigated samples were measured by both the simultaneous destructive method developed (Figure 1) and laser ablation ICP-SFMS technique. The absolute detection limits of of $^{239}Pu$ and $^{240}Pu$ determination by the destructive method (on the basis of three times the standard deviation of method blank) calculated for 10 milligram of sample amount for easier comparison are 6.0 fg ($1.4 \times 10^{-2}$ mBq) and 4.2 fg ($3.5 \times 10^{-2}$ mBq), respectively. Using higher pre-concentration factor for the Pu-measurements detection limit can be further improved. By the laser ablation ICP-SFMS the detection limit is higher due to the $^{238}U$ interference.

The $^{239}Pu$ and $^{240}Pu$ concentration in each sample investigated were below the detection limits with the exception of the Round Robin sample (RR-HEU). In this sample the measured $^{240}Pu/^{239}Pu$ was 0.279 ± 0.012 verifying the previous irradiation or mixture with reprocessed reactor-grade fuel. Though plutonium concentrations in the confiscated illicit uranium-

oxide samples were undetectable, the minor, long-lived uranium isotope, $^{236}U$ was clearly present in HU-DEP and HU-LEU samples. This finding indicates that $^{236}U$ is a better indicator for the detection of irradiation or reprocessing of uranium materials than the presence of $^{239}Pu$, as Pu/U separation is usually more effective and less sample amount is necessary for the analysis. However, the Pu isotopic composition gives useful information on the irradiation (e.g. burn-up) if it can be measured in the nuclear material.

## 4. Conclusions

A simple and rapid non-destructive has been developed for the determination of production date and Pu-concentration of uranium-based nuclear materials by isotope dilution ICP-SFMS. The method due to its low detection capabilities can be used for the age determination even if limited sample amount is available (for example, approximately 10 mg of natural $UO_2$ is usually sufficient for age determination with suitable uncertainty) or for low-enriched uranium materials. The measurement can be accomplished within two days, thus significantly faster than gamma or alpha spectrometry.

The production date can also be determined by direct laser ablation ICP-SFMS technique, which has the great advantage that it does not require the dissolution of the sample and only a small portion (approximately a few micrograms) is consumed for the analysis. However, care has to be taken to avoid the occurring molecular interferences. Though the precision of laser ablation ICP-SFMS technique is inferior to that of the liquid sample introduction, the uncertainty of the measured production date is usually adequate for nuclear forensic purposes.

The Pu-content in the uranium-oxide sample can also be measured simultaneously, however, $^{236}U$ was found to be a better indicator of previous irradiation.

## 5. Acknowledgements

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# A Brief Sensitivity Analysis for the GIRM and Other Related Technique using a One-Group Cross Section Library for Graphite-Moderated Reactors

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***Abstract:***

*Several methods have been developed previously for estimating cumulative energy production and plutonium production from graphite-moderated reactors. These methods generally fall into the category of nuclear archaeology. The Graphite Isotope Ratio Method (GIRM) is one well-known technique of this type. This method is based on the measurement of trace isotopes in the reactor's graphite matrix to determine the change in their isotopic ratios due to burnup. These measurements are then coupled with reactor calculations to determine the total plutonium and energy production of the reactor.*

*To facilitate sensitivity analysis of these methods, a one-group cross section and fission product yield library for the fuel and graphite activation products has been developed for MAGNOX-style reactors. This library is intended for use in the ORIGEN computer code which calculates the buildup, decay, and processing of radioactive materials. The library was developed using a fuel cell model in Monteburns. This model consisted of a single fuel rod including natural uranium metal fuel, magnesium oxide (magnox) cladding, carbon dioxide coolant, and Grade A United Kingdom (UK) graphite. Using this library a complete sensitivity analysis can be performed for GIRM and other techniques. The brief sensitivity analysis conducted in this study assessed various input parameters including $^{235}U$ and $^{238}U$ cross section values, aluminum alloy concentration in the fuel, and initial concentrations of trace elements in the graphite moderator. The results of the analysis yield insight into the GIRM method and the isotopic ratios the method uses as well as the level of uncertainty that may be found in the system results.*

**Keywords:** GIRM; plutonium production estimates; magnox reactors; nuclear archaeology

## 1. Introduction

Verifying the production of fissile material in nuclear facilities is a key element in the efforts of nuclear nonproliferation. The production of fissile material can result from several processes. Some of these processes include the enrichment of uranium for fuel to power nuclear reactors, the production of plutonium within the fuel of a reactor during operation, and the buildup of $^{233}U$ from $^{232}Th$ via the irradiation of thorium. During these processes some evidence is left behind that could lead an investigator to predict the most likely events from the past that would have led to the evidence observed in the present day. This study of evidence and its relationship to past material production is often known as "nuclear archaeology" and is rooted in the verification of nuclear weapons activities [1].

Several methods have been developed previously for estimating cumulative energy production and plutonium production from graphite-moderated reactors. The Graphite Isotope Ratio Method (GIRM) is one well-known technique. The GIRM technique was originally developed as a joint venture between the United States and Russia as a method to evaluate the large uncertainty associated with the Russian plutonium production during the history of its weapons program [2]. While the main focus of

the work was Russian graphite-moderated reactors, this method is applicable to any graphite-moderated reactor. Reactors of this type have been operational in several countries throughout the world including Great Britain, France, the United States, Russia, and Japan [3].

### 1.1. Description of GIRM

The basic nuclear physics principle of GIRM is that atoms undergo predictable changes during neutron irradiation. The neutron fluence in a reactor is defined as the time integral of the neutron scalar flux. The evaluation of the changes caused by the exposure of a neutron fluence to the natural uranium fuel, graphite moderator, or other structural materials provides some measure of the fluence that caused those changes. The energy produced in the core of the reactor is a direct measure of the number of fission reactions that have occurred. For each fission reaction approximately 2.4 neutrons are produced. Of these neutrons, one must be absorbed in $^{235}U$ to cause a subsequent fission to maintain the critical chain reaction. Some fraction of the remaining neutrons is absorbed in $^{238}U$ to produce plutonium, and some other fraction is absorbed in other materials such as the moderator and structural materials. The absorption of neutrons by the trace elements found in the graphite moderator causes shifts in the isotopic composition of those trace elements. Modern reactor physics codes such as WIMS, MCNP, DANT, and 1DB can accurately predict the behavior of the neutrons and the fractions of neutrons absorbed in the various structural components of the reactor.

The GIRM method is useful as a verification tool for the plutonium production of graphite-moderated reactors. Impurities in the graphite come from the environment when graphite ore is mined from the earth. Even with concentrations at parts per million levels, the graphite impurities are measurable with mass spectroscopy. These mass measurements are the first step in GIRM. Several samples are drilled from a range of locations in the reactor core using commercially available machinery. Locations of the samples can be at nearly any axial location in the core, from any fuel channel, and at any depth into the graphite. The quantity and location of these samples are optimized for each reactor.

The next step of GIRM involves fuel cell calculations using a reactor physics code. A fuel cell model consists of one fuel rod and the surrounding materials in the fuel channel which include the carbon dioxide coolant and graphite moderator. The fuel cell calculation provides the uranium fuel rod burnup in terms of megawatt-days per kilogram of fuel (MWd/kg), which is the time integrated reactor power per unit mass of fuel. This computation also generates the relationship between the fuel isotopes and the trace isotopes in the graphite that are immediately adjacent to the fuel rod. Using this correlation, the measured ratio from a graphite sample, for instance B-10/B-11, can estimate the energy and plutonium produced at the specific location associated with the sample [4]. The result is a set of local plutonium production estimates throughout the reactor.

The last step in GIRM includes using the set of local fuel burnup values with a regression analysis technique to fit a series of basis functions to the measured data. The result is a three-dimensional, full-core fluence profile which best fits the set of local estimates and is defined for every point in the reactor. The fluence field is assumed to be a linear combination of eigenfunctions. When detailed operational data is unknown, these functions are found to be the eigenfunction solutions to the homogeneous diffusion equation for the core [4]. Once an adequate weighted regression of these functions is developed for the fluence model, the average production of plutonium in the reactor is determined from the integral of the fluence over the reactor volume.

## 2. One-group cross section library

A one-group cross section and fission product yield library was previously developed for graphite-moderated reactors and is intended for use with the ORIGEN code [5]. This library was based on the Calder Hall reactor design. The cross section library was successful in calculating uranium and plutonium concentrations for average burnups of 3000 to 4000 MWD/MTU in graphite-moderated reactors. At higher burnups of 5000 and 6000 MWD/MTU, the library resulted in $^{235}U$ concentrations within 4% and $^{239}Pu$ concentrations within 2%. $^{238}Pu$ and $^{240}Pu$ had errors of up to 10% in some cases at these higher burnups.

The ORIGEN code has both advantages and disadvantages in its use for a sensitivity analysis. The most important advantage to the code's use in a sensitivity analysis is the speed at which it operates.

A single simulation executes on the order of seconds to minutes. This accommodates numerous variations in the input parameters. In addition to its rapid execution, the cross section libraries used with ORIGEN are easily interpreted text documents. It is an easy task for the user to make changes in the cross section library to analyze fluctuations and uncertainties associated with the result of the GIRM method due to the cross section values.

## 3. Sensitivity analysis

For use in safeguards analysis and materials accountability, it is often the case that not every detail of the operating history of the reactor or every parameter of the material is perfectly known or reliable. It is, therefore, important to know which factors and parameters have the greatest impact on the answers that are calculated with various techniques. This information is most often obtained from a thorough sensitivity analysis.

A sensitivity analysis is useful in finding the specific parametric sources of error and to quantify the level of error that uncertainty in these specific parameters can cause in the results of GIRM. Using the library developed for the ORIGEN code, several cases were run to analyze the effect of small changes in fuel and graphite input parameters. These analyses are:

- the effect of the initial weight percent of aluminum alloy in the fuel on the uranium and plutonium isotopic ratios at the end of burnup
- the change in uranium and plutonium isotopic ratios at the end of burnup due to ±1% and ±5% changes in the uranium and plutonium cross section values used in the calculations
- the dependence at a specified burnup of the isotopic ratios of trace elements in the graphite to each element's initial concentration

### 3.1. Aluminum alloy concentration

Most uranium metal fuels are an alloy, and uranium is often alloyed with small percentages of aluminum. In this analysis, the aluminum alloy percentage was altered to simulate uncertainty in the initial fuel composition. A simulation with no aluminum in the fuel was used as the basis for comparison to cases with 0.5, 1.0, and 2.0 w/o aluminum in the fuel. Figure 1 displays the percent difference of the uranium and plutonium isotopes at a burnup of 3093 MWD/MTU as compared to the case with no aluminum alloy in the fuel.

The change in the uranium results is approximately the same as the percentage of initial aluminum in the fuel. Pu-239 follows similar behavior but is less than the original result by approximately half the weight percent concentration of aluminum. For example, in the simulation with 1.0 w/o Al, the Pu-239 is 0.41% below the result without any alloy which is approximately half the magnitude of the change in aluminum alloy weight percent found initially in the fuel. The other plutonium isotopes, however, overestimate the EOI concentrations as compared to the case without aluminum.

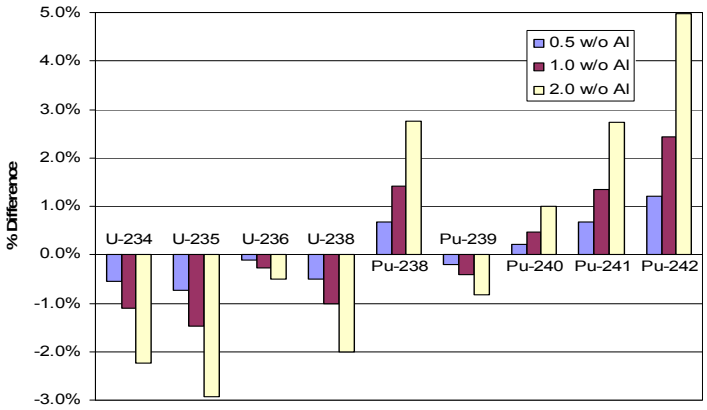


**Figure 1:** Difference of ORIGEN results compared to case with no aluminum alloy in fuel

### 3.2. Uranium cross sections

The sensitivity of the uranium cross section data was next analyzed. Neutron cross sections are measured quantities and thus have some associated error. Analyzing fluctuations in the U-235 and U-238 neutron cross section values can help quantify the magnitude of the effects from this uncertainty.

Both the U-235 and U-238 cross section data in the ORIGEN library were changed by ±1% and ±5% for a total of ten different scenarios. The results of these cases determine the effect of the cross section fluctuations on the uranium and plutonium measurements at the end of irradiation. The results for these parameter changes are shown in Table 1 and Table 2 for the nuclides of interest. Table 1 displays the results at the end of irradiation when the input cross section data for U-235 was changed. Table 2 displays the results at the end of irradiation when the input cross section data for U-238 was changed. In the tables, "XS" is used to represent "cross-section".

| | Original | +5% U-235 XS Change | | +1% U-235 XS Change | | -1% U-235 XS Change | | -5% U-235 XS Change | |
|---|---|---|---|---|---|---|---|---|---|
| **Nuclide** | **[g/MTU]** | **[g/MTU]** | **Error** | **[g/MTU]** | **Error** | **[g/MTU]** | **Error** | **[g/MTU]** | **Error** |
| **U-235** | 4.436E+03 | 4.388E+03 | -1.08% | 4.426E+03 | -0.23% | 4.445E+03 | 0.20% | 4.486E+03 | 1.13% |
| **U-238** | 9.896E+05 | 9.897E+05 | 0.01% | 9.896E+05 | 0.00% | 9.896E+05 | 0.00% | 9.895E+05 | -0.01% |
| **Pu-238** | 1.195E+00 | 1.129E+00 | -5.52% | 1.181E+00 | -1.17% | 1.210E+00 | 1.26% | 1.268E+00 | 6.11% |
| **Pu-239** | 1.960E+03 | 1.929E+03 | -1.58% | 1.954E+03 | -0.31% | 1.966E+03 | 0.31% | 1.992E+03 | 1.63% |
| **Pu-240** | 2.723E+02 | 2.615E+02 | -3.97% | 2.700E+02 | -0.84% | 2.745E+02 | 0.81% | 2.837E+02 | 4.19% |

**Table 1:** Comparison of ORIGEN results at a burnup of 3093 MWD/MTU for changes in the U-235 cross section

| | Original | +5% U-238 XS Change | | +1% U-238 XS Change | | -1% U-238 XS Change | | -5% U-238 XS Change | |
|---|---|---|---|---|---|---|---|---|---|
| **Nuclide** | **[g/MTU]** | **[g/MTU]** | **Error** | **[g/MTU]** | **Error** | **[g/MTU]** | **Error** | **[g/MTU]** | **Error** |
| **U-235** | 4.436E+03 | 4.466E+03 | 0.68% | 4.442E+03 | 0.14% | 4.429E+03 | -0.16% | 4.404E+03 | -0.72% |
| **U-238** | 9.896E+05 | 9.895E+05 | -0.01% | 9.896E+05 | 0.00% | 9.896E+05 | 0.00% | 9.897E+05 | 0.01% |
| **Pu-238** | 1.195E+00 | 1.204E+00 | 0.75% | 1.197E+00 | 0.17% | 1.194E+00 | -0.08% | 1.186E+00 | -0.75% |
| **Pu-239** | 1.960E+03 | 2.041E+03 | 4.13% | 1.976E+03 | 0.82% | 1.944E+03 | -0.82% | 1.879E+03 | -4.13% |
| **Pu-240** | 2.723E+02 | 2.796E+02 | 2.68% | 2.737E+02 | 0.51% | 2.708E+02 | -0.55% | 2.646E+02 | -2.83% |

**Table 2:** Comparison of ORIGEN results at a burnup of 3093 MWD/MTU for changes in the U-238 cross section

Figure 2 graphically displays the change in the concentration of Pu-239 and Pu-240 as a function of variation in the U-235 and U-238 cross section values. All data is for a burnup of 3093 MWD/MTU. The calculated concentrations of the uranium and plutonium isotopes above are linearly related to the value of the U-235 and U-238 cross sections used in the ORIGEN library. The equations displayed on each figure are the best fit straight line for the data. The $R^2$ value indicates how well the trend line fits the data, and the trend line is most reliable when the $R^2$ value is near one. The slope of the trend lines are indicative of the magnitude of the effect that the cross sections have on the calculated concentrations at the 3093 MWD/MTU burnup. This value is also useful for the propagation of error in the results of GIRM due to a quantitative uncertainty in the cross section.

The Pu-238 is insensitive to the U-238 cross section because it is produced from U-235 and not U-238. The Pu-239, however, is much more sensitive to changes in the U-238 cross section since it is produced from U-238. The figures below show the linear dependence of the plutonium isotopic concentrations to the values of the U-235 and U-238 cross sections. The change of the absorption cross sections directly affects the plutonium buildup and depletion. This sensitivity is largest for the plutonium isotopes that are closest in mass to the uranium isotopes and produced first in the plutonium buildup chain.

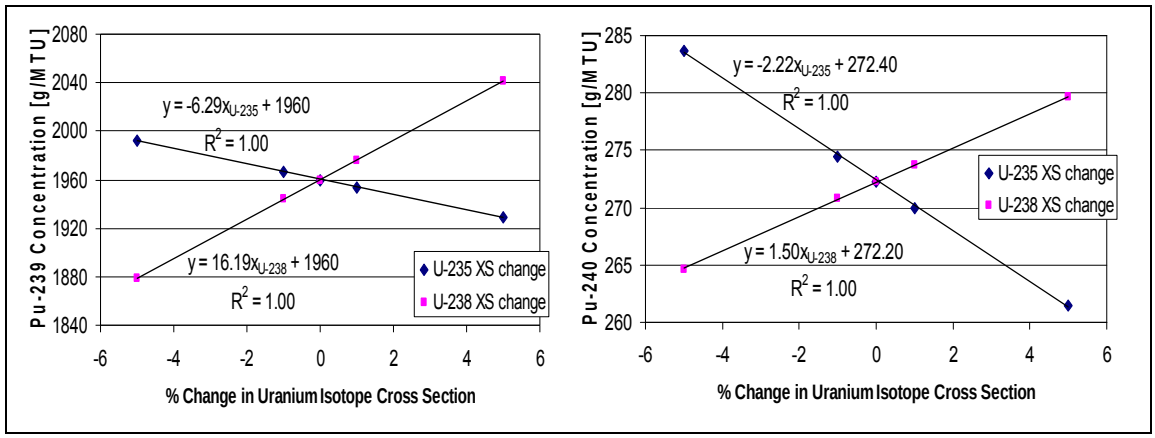


**Figure 2:** Effect of uranium cross sections on the U-235, U-238, Pu-239, and Pu-240 concentrations at a fuel burnup of 3093 MWD/MTU

### 3.3 Impurity concentrations in graphite

For each of the graphite impurities, ORIGEN decks were run with the concentration varied one element at a time: the initial amount of each impurity found in grade A UK graphite, one-fourth, one-half and double the initial amount. The initial concentrations from grade A UK graphite (represented by N(t=0) below) are 6.396E-08, 3.642E-04, and 9.994E-10 g/kg-graphite for Ti, Li, and Cl, respectively.

The $^{48}Ti/^{49}Ti$ ratio was chosen as an indicator ratio in the development of GIRM for intermediate to high fluence ranges. For changes in the titanium initial concentration, Table 3 shows the results and percent change of the results as compared to the model with the standard UK graphite impurity concentration. The Ti-46/Ti-48 ratio was the least sensitive to change in initial concentration as it varied by less than 1% for any change from the standard initial concentration. The largest change in ratios came after quartering the initial standard concentration of titanium and resulted in changes of approximately 19% and 24%. The ratios of Ti-47/Ti-48, Ti-49/Ti-48, and Ti-50/Ti-48 decreased as the amount of initial titanium increased. The trend of this change can be seen in Figure 3 where the standard initial titanium concentration was 6.396E-08 grams per kilogram of graphite.

A competing (n,α) reaction produces titanium from chromium. When the concentration of titanium is small enough, this reaction dominates the titanium production in the graphite. Thus, the titanium concentration is not solely dependent on the neutron absorptions in its own isotopes. This explains the increasing change in the isotope ratios as the initial concentration is decreased. The results suggest that the Ti-46/Ti-48 ratio could be used in the GIRM analysis since it is insensitive to the amount of titanium found in the graphite prior to irradiation. However, the Ti-46/Ti-48 ratio would not be effective as an indicator ratio since it does not have a strong correlation to neutron fluence remaining nearly constant throughout the operation of the reactor.

| | N(t=0) | 0.25*N(t=0) | % Change | 0.5*N(t=0) | % Change | 2*N(t=0) | % Change |
|---|---|---|---|---|---|---|---|
| **Ti-46/48** | 0.1080 | 0.1081 | 0.10% | 0.1081 | 0.06% | 0.1080 | 0.03% |
| **Ti-47/48** | 0.1082 | 0.1345 | 24.29% | 0.1169 | 8.09% | 0.1038 | -4.03% |
| **Ti-49/48** | 0.0885 | 0.1054 | 19.00% | 0.0942 | 6.33% | 0.0858 | -3.13% |
| **Ti-50/48** | 0.0789 | 0.0941 | 19.24% | 0.0840 | 6.43% | 0.0764 | -3.18% |

**Table 3:** ORIGEN results for titanium isotopic ratios in the graphite with varying initial titanium concentrations for a fuel burnup of 3093 MWD/MTU

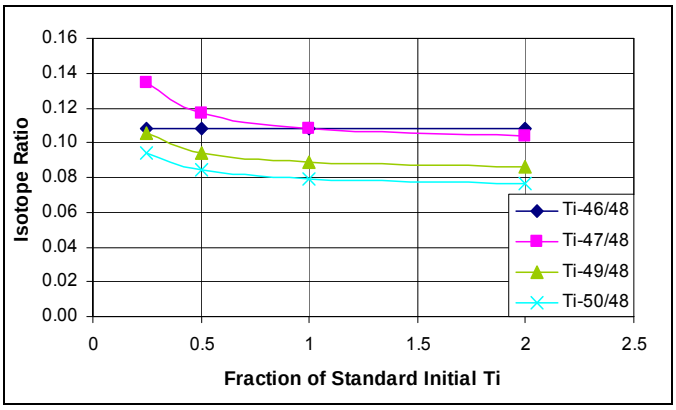


**Figure 3:** Titanium isotopic ratios in the graphite as a function of initial titanium concentration at a fuel burnup of 3093 MWD/MTU

Another indicator ratio in the GIRM method is the Li-6/Li-7 isotopic ratio. Both Li-6 and Li-7 are the only two isotopes of lithium found in nature, and all other isotopes of lithium have half-lives of less than one second. As compared to the titanium analysis which had almost a 25 percent change for one isotopic ratio, the Li-6/Li-7 ratio results in Table 4 do not demonstrate a huge variation (less than two percent for all cases). On the other hand, the curve of the data which Figure 4 displays indicates that the relationship of the isotopic ratio to the initial lithium concentration is not linear and could grow increasingly worse with less and less initial lithium concentration. To avoid this, the lithium could be disregarded as an indicator element if its concentration is small enough that it is near the limits of measurability.

| | N(t=0) | 0.25*N(t=0) | % Change | 0.5*N(t=0) | % Change | 2*N(t=0) | % Change |
|---|---|---|---|---|---|---|---|
| **Li-6/Li-7** | 0.0202 | 0.0199 | -1.74% | 0.0201 | -0.55% | 0.0203 | 0.25% |

**Table 4:** ORIGEN results for lithium isotope ratio in the graphite with varying initial lithium concentrations for a fuel burnup of 3093 MWD/MTU

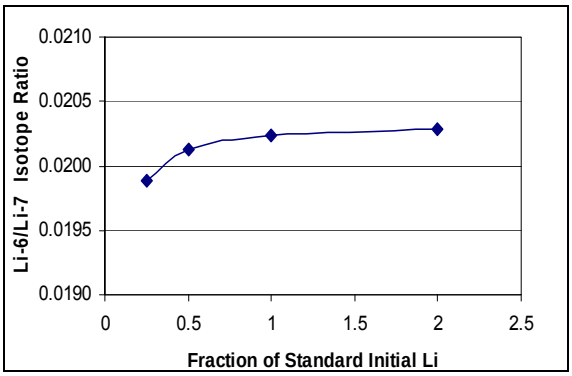


**Figure 4:** Li-6/Li-7 isotopic ratio in the graphite as a function of initial lithium concentration at a fuel burnup of 3093 MWD/MTU

The result of the analysis of for the initial chlorine concentration in Table 5 showed that the largest change came from the ratio of chlorine's natural isotopes, Cl-37/Cl-35, which varied at the greatest by 10%. The Cl-36/Cl-35 ratio did change by almost 5% when the initial amount of chlorine was dropped to a concentration of 0.25*N(t=0). The Cl-37/Cl-36 ratio demonstrated errors as large as 16% and is

thus unsuitable for the GIRM method. The trend in the Cl-36/Cl-35 and Cl-37/Cl-35 changes can be seen in Figure 5.

| | N(t=0) | 0.25*N(t=0) | % Change | 0.5*N(t=0) | % Change | 2*N(t=0) | % Change |
|---|---|---|---|---|---|---|---|
| **Cl-36/35** | 0.042 | 0.040 | -4.70% | 0.042 | -1.67% | 0.043 | 0.88% |
| **Cl-37/35** | 0.3672 | 0.4068 | 10.78% | 0.3814 | 3.86% | 0.3599 | -1.99% |
| **Cl-37/36** | 8.69 | 10.1 | 16.25% | 9.18 | 5.62% | 8.45 | -2.84% |

**Table 5:** ORIGEN results for chlorine isotopic ratios in the graphite with varying initial chlorine concentrations for a fuel burnup of 3093 MWD/MTU

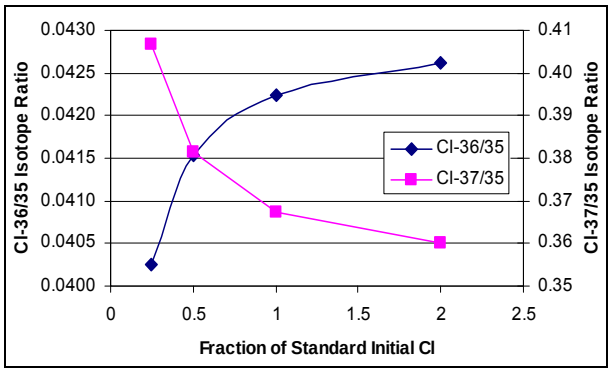


**Figure 5:** Chlorine isotopic ratios in the graphite as a function of initial lithium concentration at a fuel burnup of 3093 MWD/MTU

The results show that the Cl-36/Cl-35 ratio had the least sensitivity of the three ratios to the initial chlorine concentration, but this sensitivity is still very large compared to the source of error from regression models and reactor physics calculations. The ratio of Cl-36/Cl-37 which had the least sensitivity of the chlorine ratios could still produce systematic errors of up to five percent. These sensitivities are due to the production of chlorine from the beta decay of S-35 and S-36. Since the initial concentration of the impurities in the graphite is never known, the data suggests that chlorine and titanium would not make suitable indicator elements in the GIRM method. Using chlorine or titanium isotopic ratios in the GIRM analysis could lead to extremely high errors in the plutonium production estimates.

## 4. Conclusion

Based on the above sensitivity analysis, the systematic errors associated with some of the uncertainties analyzed here are much greater than the errors previously studied. An earlier error analysis of the analysis step in GIRM studied the error associated with the reactor physics calculations and regression model [6]. The conclusion was an error of 1.62% on the plutonium estimation. This error can be minimized by increasing the number of sample taken from the graphite. However, the systematic errors seen from the above parameters cannot be reduced by any such method.

For instance, the large errors due to changes in the graphite impurity concentrations are unavoidable if the initial impurity concentration is not known. During the application of GIRM for a comparative plutonium estimate in the Trawsfynydd reactor, the analysis of graphite samples concluded that the titanium impurities in the graphite had a substantial heterogeneity throughout the matrix. Additionally, a significant amount of variance in the titanium ratios was seen beyond the expected errors [7]. Thus, chlorine and titanium should not be used as indicator elements in GIRM due to the large sensitivities of their isotope ratios to the initial concentration of the element. Errors in the system results due to the other input values such as the aluminum alloy concentration and uranium cross sections should be recognized and considered in the estimates from GIRM.

The ORIGEN cross section and fission yield library developed in these studies can be applied to any graphite-moderated reactor. The ORIGEN code provides a time-efficient method for a sensitivity analysis and burnup and depletion calculations. The ORIGEN code itself is also advantageous in its simplicity and ease in manipulating cross section data and its ability to change power and neutron flux within the system. However, the ORIGEN code has certain disadvantages. It does not use pointwise neutron cross section data to account for energy dependence of the neutrons. The code also does not possess the capability to accommodate changes in geometry, and the user can only change the system power or neutron flux. Analyses involving various sensitivities to geometry require a more complex reactor physics code which will result in longer computational times. However, these more complex codes can provide a computational analysis of sensitivities that include the dependence of a particular isotopic ratio to the location in the graphite matrix.

Additional sensitivity analyses could be performed to determine similar relationships for other isotopes and parametric studies. These relationships provide a basis for the type of input data required to obtain accurate answers using such techniques as GIRM and the error than is associated with its results. These material verifications are essential in material protection, control, and accountability and the assurance of global security.

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# Advancing Methods for Providing Attribution for the HEU Used in a Terrorist Nuclear Weapon

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***Abstract:***

*An algorithm was developed that uses measured isotopic ratios from fission product residue following the detonation of a high-enriched uranium nuclear weapon to compute the original attributes of the nuclear material used in the device. The specific attributes assessed are the uranium isotopics (considering $^{234}U$, $^{235}U$, $^{236}U$, and $^{238}U$) and the type of enrichment process used to create the material (e.g., gaseous diffusion, gas centrifuge, etc.). Using the original material attributes of the weapon significantly increases the probability of identifying the perpetrator of the attack. In this study, research was conducted to perform sensitivity analysis of the calculated values, analyze alternate methods of enrichment, determine the source (uranium mine) from which the feed material was taken and assess potential "spoofing" techniques. The purpose of this research was to verify that the analytical method developed would remain valid for a multitude of conceivable variations that could potentially be used to disguise the origin of the original nuclear material used in the device. It is envisioned that this methodology could serve as a pre-processing step to a more computationally intensive and more accurate system in the event of a nuclear terrorist attack.*

**Keywords:** highly enriched uranium; material attribution; terrorism; post-detonation analysis;

## 1. Introduction

The development and implementation of safeguards to prevent a nuclear terrorist attack is crucial to improving security throughout the global community. If a terrorist nuclear device was detonated, how quickly could the site be assessed to determine what type of device was detonated, how powerful the device was and where it came from? The detonation of an HEU weapon would cause catastrophic damage and mass casualties. Due to the severity of such an attack, it is critical to be able to compute the original material attributes of a weapon because it significantly increases the probability of identifying perpetrators of the attack.

The objective of the algorithm developed was to utilize post-detonation measured isotopic ratios in order to determine the pre-detonation material attributes within reasonable accuracy. More computationally intensive methods are being developed elsewhere; however, these methods require extensive computational times in order to produce acceptable results. In effort to reduce the computational time required to compute the original material attributes, the method developed here uses an analytical approach which consisted inversions of the burnup and decay equations (all first-order ordinary differential equations). It is envisioned that this methodology could serve as a pre-processor step to a more computationally intensive and more accurate system.

This work is focused on the post-detonation attribution of a Highly Enriched Uranium (HEU) terrorist nuclear weapon. Terrorist devices may differ from military nuclear weapons mainly in the sophistication

applied when constructed (e.g. type and grade of material used and quality of tamper/reflector). Since a gun-type weapon is considerably less complex than an implosion weapon (generally, gun-type weapons are not tested), this is considered to be a likely scenario for a nuclear terrorist attack [1].

Given a measurement of the post-detonation isotopics from fission product residue, the interest in this work was to attempt to determine the following characteristics (in this order of importance): (1) pre-detonation $^{235}U$ enrichment, (2) pre-detonation $^{234}U/^{238}U$ isotopic ratio, (3) pre-detonation $^{236}U/^{238}U$ isotopic ratio, (4) enrichment method used to produce material, (5) pre-enrichment $^{234}U/^{238}U$ isotopic ratio, (6) pre-enrichment $^{236}U/^{238}U$ isotopic ratio, and (7) source (mine or otherwise) from which feed uranium was taken. It was acknowledged immediately that steps (1)-(3) would have a likely chance of success and the steps (4)-(7) would be significantly more difficult.

## 2. Methodology

The algorithm developed here consists of two main parts: a forward model and an inverse model. The forward model consisted of simulations to predict post-detonation (actually post-irradiation) isotopics given the original isotopics of the material and the number of fission (or yield) of the device. The data from the forward model was mainly used to test the viability of the inverse model. The inverse model predicted pre-detonation isotopics using analytical inversions of the buildup and decay equations and post-detonation isotopic measurements. The inverse model also included error propagations to allow for prediction of uncertainties in the attributes as well as to determine the sensitivity of the results to the input data.

### 2.1. Forward model

The forward model simulations were performed using the ORIGEN2 computer code [2]. ORIGEN2 calculates the buildup and depletion of isotopics from irradiation and decay. The code possesses a large set of libraries (each library corresponds to a specific type of reactor) with cross-section, decay, and fission product yield data. ORIGEN2 uses the matrix exponential method to solve a large system of coupled, linear, first-order ordinary differential equations. While not a weapons burn code, ORIGEN contains sufficient capability to allow for analysis of the feasibility of the method developed here.

Four different uranium signatures from gaseous centrifuge and gaseous diffusion enriched uranium, both with and without $^{236}U$ present in the original material, were simulated. In order to simulate the detonation of a 20 kT HEU weapon in ORIGEN2, the mass of $^{234}U$, $^{235}U$, $^{236}U$ (if applicable), and $^{238}U$ were calculated. Assuming the total mass of uranium equalled one metric ton and was enriched to 95 a/o $^{235}U$, values for the enrichment of $^{234}U$ and $^{236}U$ were calculated*. Natural uranium contains essentially no $^{236}U$ (though small quantities are found in natural material due to the activation of $^{235}U$ from neutron background); however, enriched uranium of U.S. or Russian origin includes a significantly higher abundance of $^{236}U$ due to the re-enrichment of naval fuel. Thus, the presence of $^{236}U$ in the original material provides a unique signature indicating the geographic origin of where the uranium was enriched.

Then, the burnup of the initial material in the weapon given a 20 kT yield was simulated using ORIGEN2. Generally, a 2 kT yield is associated with terrorist weapons; however, this value was not used because only 2% of the original material fissions. The task of determining the original material used in the weapon becomes much simpler for low yields because there is only a slight difference between the pre-detonation composition and the post-detonation composition of the weapon. The resultant isotopics produced from this burnup were then decayed for 1.0 day (assumes that it will take approximately 1 day or more to acquire measurements from the post-detonation fission product residue). Assuming that the weapon was detonated on the ground or at a relatively low altitude, $^{89}Sr$ and $^{95}Zr$ (characterized by long half-lives, low absorption cross-sections, and the ability to be measured in the environment) were the two fission products used to calculate the total number of fission from the device in the inverse model.

* Equations used in algorithm may be obtained from the original paper (reference 3).

### 2.2. Inverse model

The inverse model equations are all expressed in terms of atom ratios relative to $^{238}U$ (the $^{238}U$ concentration in the device is roughly constant during irradiation). The algorithm [3] implemented in the inverse model uses an iterative procedure to calculate the original material attributes* which consists of the following steps:

1. The pre-detonation $^{235}U/^{238}U$ ratio is set to an initial guess input by the user.
2. A guess was made for the method of enrichment used and whether or not $^{236}U$ was present in the initial material and the corresponding pre-detonation $^{234}U/^{238}U$ and $^{236}U/^{238}U$ (if applicable) ratios were calculated using eqs. (2.1) - (2.3).
3. The pre-detonation $^{234}U/^{238}U$ and $^{236}U/^{238}U$ (if applicable) ratios were combined with the initial guess for $^{235}U/^{238}U$ to calculate the $^{235}U$ enrichment of the original material.
4. The number of fissions in the device per unit mass was calculated using the measurement of two fission products: $^{95}Zr$ and $^{89}Sr$.
   - A single fission product could have been used but by using two fission products, iteration between the two yielded a better prediction of the number of fissions.
   - The equation derived for the total number of fissions assumed that the recoverable energy per fission from $^{235}U$ equalled 200 MeV and that all fissions were from $^{235}U$.
5. An updated $^{234}U/^{238}U$ value was calculated using measurements of $^{232}U/^{238}U$ in the residue.
   - It was assumed that no $^{232}U$ existed in the original material and the measured $^{232}U$ concentration was produced only from the $^{234}U(n,3n)^{232}U$ reaction.
6. An updated $^{235}U/^{238}U$ value was then calculated using measurements of $^{235}U/^{238}U$ in the residue.
   - It was assumed that the change in $^{235}U$ was equal to its loss rate from absorption.
7. Then, an updated $^{236}U/^{238}U$ value was calculated using measurements of $^{236}U/^{238}U$ in the residue.
   - This derivation assumed that the change in $^{236}U$ was equal to its production rate from radiative capture in $^{235}U$ minus the loss rate from the absorption of $^{236}U$.
   - The final equation for the updated $^{236}U/^{238}U$ value was obtained by assuming that the ratio of $^{236}U/^{235}U$ as a function of irradiation time was linear and therefore was easily integrated.
8. A new value for the $^{235}U$ enrichment was then calculated and steps (4) – (7) were repeated iteratively until the pre-detonation $^{235}U/^{238}U$ ratio converged to a value within a specified tolerance.

## 3. Uranium Signatures

### 3.1. Enrichment processes

Weapons-grade HEU is typically enriched to 90 a/o $^{235}U$ or greater. The method of enrichment provides a useful signature that may indicate where the uranium was enriched. Methods used to enrich uranium include: gaseous centrifuge, gaseous diffusion, electromagnetic isotope separation (EMIS), and atomic vapor laser isotope separation (AVLIS) [1]. The two most common enrichment processes used throughout the world are gaseous centrifuge and gaseous diffusion both of which separate the uranium isotopes in a gaseous compound called uranium hexafluoride.

In this study, the algorithm developed analyzed only gaseous centrifuge and gaseous diffusion enrichment methods. These methods are hard to distinguish because they both rely on the differences in mass between $^{235}U$ containing molecules and $^{238}U$ containing molecules, though they are based on different physical processes. This results in small separation factors of 1.162 and 1.00429 for gaseous centrifuge and gaseous diffusion, respectively, due to the higher concentration of $^{234}U$ contamination in the enriched product [4]. It is expected that distinguishing most other methods (such as AVLIS or EMIS) would be much simpler. For instance, the AVLIS process enriches uranium using lasers tuned to a precise frequency so that only the $^{235}U$ atoms absorb the light. The resulting separation factor is nearly infinite and yields almost no $^{234}U$ in the enriched product.

In order to determine valid signatures indicating the method of enrichment, the values calculated in the inverse model for post-detonation $^{234}U$ concentrations were compared. For 95 a/o $^{235}U$ centrifuge enriched

fuel, the calculated $^{234}U/^{238}U$ ratio was approximately 5.0 times greater than the calculated $^{234}U/^{238}U$ ratio for 95 a/o $^{235}U$ diffusion enriched fuel. These significant variations in $^{234}U$ are presented in Table 1 and were used as signatures indicating the enrichment process used.

| Enrichment Process | | Pre-detonation Value $(N^{234}/N^{238})_0$ | Inverse Model $(N^{234}/N^{238})_0$ | Percent Difference |
|---|---|---|---|---|
| Centrifuge | with $^{236}U$ | 1.04 | 1.06 ± 0.015 | 1.66% |
| | without $^{236}U$ | 0.869 | 0.883 ± 0.013 | 1.69% |
| Diffusion | with $^{236}U$ | 0.200 | 0.204 ± 0.003 | 2.20% |
| | without $^{236}U$ | 0.179 | 0.182 ± 0.005 | 1.31% |

**Table 1:** Comparison of calculated $^{234}U/^{238}U$ ratios to distinguish centrifuge enriched fuel from diffusion enriched fuel.

### 3.2. Presence of $^{236}U$

After the enrichment process has been determined, whether or not $^{236}U$ existed in original weapons material must be established. The $^{236}U/^{238}U$ inverse model values computed for gaseous diffusion and gaseous centrifuge enriched uranium, both with and without $^{236}U$ present, are presented in Table 2. For enriched fuel with $^{236}U$ present in the original material, the calculated $^{236}U/^{238}U$ value was approximately 4.5 times greater than the $^{236}U/^{238}U$ value for enriched fuel without $^{236}U$ present in original material.

| Enrichment Process | | Pre-detonation Value $(N^{236}/N^{238})_0$ | Inverse Model $(N^{236}/N^{238})_0$ | Percent Difference |
|---|---|---|---|---|
| Centrifuge | with $^{236}U$ | 0.195 | 0.204 ± 0.011 | 4.58% |
| | without $^{236}U$ | 0.0 | 0.005 ± 0.008 | - |
| Diffusion | with $^{236}U$ | 0.115 | 0.121 ± 0.007 | 5.57% |
| | without $^{236}U$ | 0.0 | 0.027 ± 0.003 | - |

**Table 2:** Comparison of calculated $^{236}U/^{238}U$ values to determine whether or not $^{236}U$ was present in original material.

## 4. Sensitivity Analysis

The methodology developed was tested for a 20 kT detonation of a 95 a/o $^{235}U$ enriched HEU device. The "measured values" were produced from ORIGEN simulations for four different uranium signatures from gaseous centrifuge and gaseous diffusion enriched uranium, both with and without $^{236}U$ present in the original material. Error propagations were done by hand to predict uncertainties in the attributes as well as to determine the sensitivity of these results to errors in the input data.

### 4.1. Sensitivity of initial guess for $^{235}U$ concentration

The algorithm was insensitive to the initial guess for $^{235}U$ concentration. In all cases less than 10 iterations (less than 1 second computational time) were used to acquire a result. The results presented in Table 3 verified that for any positive initial guess of any order of magnitude input into the algorithm will be iterated to a reasonably correct answer.

| Enrichment Process | Initial Guess $(N^{235}/N^{238})_0$ | Pre-detonation Value $(N^{235}/N^{238})_0$ | Inverse Model $(N^{235}/N^{238})_0$ | Percent Error |
|---|---|---|---|---|
| Centrifuge (with $^{236}U$) | $1.00 \times 10^{10}$ | 42.4 | 43.1 ± 0.431 | 1.61% |
| Diffusion (no $^{236}U$) | $1.00 \times 10^{-10}$ | 22.4 | 22.6 ± 0.225 | 0.66% |

**Table 3:** Comparison of calculated $^{235}U/^{238}U$ values from inverse model to actual values for various initial guesses.

## 4.2. Sensitivity of error in calculated $^{234}U$ attribute

Error propagations were used to derive an equation for the error in the $^{234}U/^{238}U$ attribute in terms of the errors in the input parameters (this method was repeated for $^{235}U/^{238}U$ and the $^{236}U/^{238}U$ attributes). Using this equation, the sensitivity of the error in the $^{234}U/^{238}U$ attribute was determined by plotting the error in the calculated $^{234}U/^{238}U$ value as a function of the error in the measured $^{232}U$ value and the $^{234}U(n, 3n)$ microscopic cross-section. The plot depicted in Fig. 1 shows that the calculated error in the $^{234}U/^{238}U$ value varies linearly as a function of the error in the measured $^{232}U$ value and the error in the $^{234}U(n, 3n)$ microscopic cross-section. The linear relationship determined is important because it indicates that error in the measured $^{232}U$ value and the error in the $^{234}U(n, 3n)$ microscopic cross-section equally contribute to overall error in the calculated the $^{234}U/^{238}U$ value. This relationship may also be utilized to determine the point at which reducing these errors no longer reduces the overall error in the calculated the $^{234}U/^{238}U$ attribute.

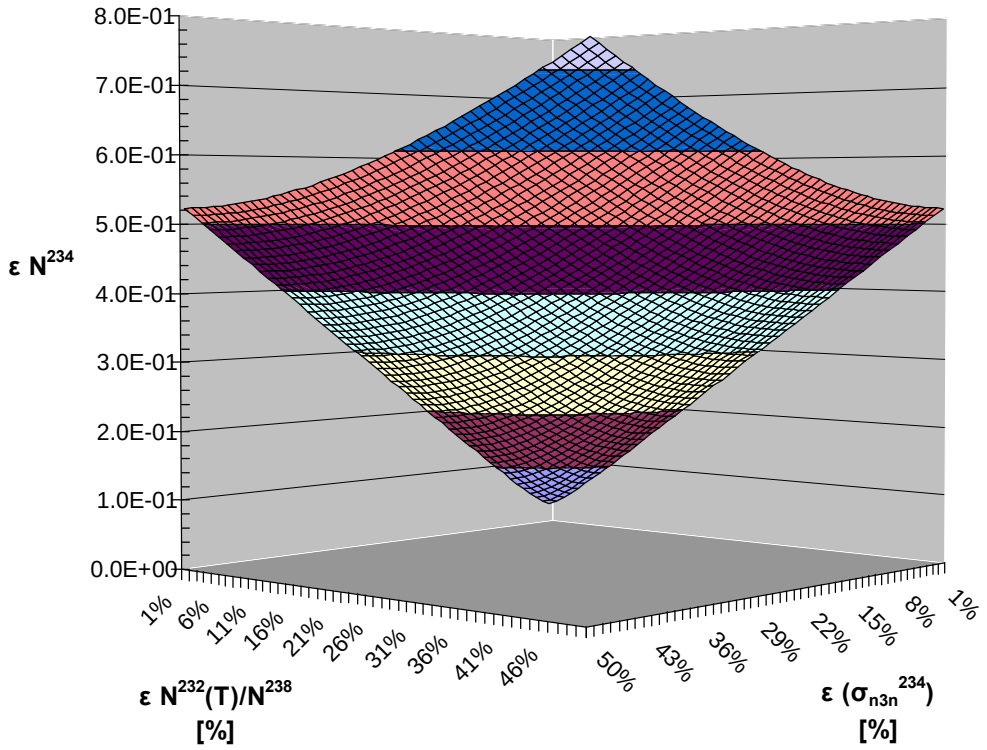


**Figure 1.** Error in the calculated $^{234}U/^{238}U$ value as a function of the error in the measured $^{232}U$ value and the $^{234}U(n, 3n)$ microscopic cross-section.

### 4.3. Sensitivity of error in calculated $^{235}U$ attribute

The error in the calculated $^{235}U/^{238}U$ attribute as a function of the error in the $^{235}U$ enrichment and the error in the $^{235}U$ microscopic fission cross-section to determine the sensitivity of the error in this attribute. The plot depicted in Fig. 2 shows that the calculated error in the $^{235}U/^{238}U$ value varies linearly as a function of the error in the $^{235}U$ microscopic fission cross-section and varies nonlinearly as a function of the error in the $^{235}U$ enrichment. The nonlinear relationship determined indicates that error in the $^{235}U$ enrichment contributes more towards the overall error in the calculated the $^{235}U/^{238}U$ value than the error in the $^{235}U$ microscopic fission cross-section does. Therefore, more effort should be spent reducing the error in the value for the $^{235}U$ enrichment than reducing the error in the $^{235}U$ microscopic fission cross-section.

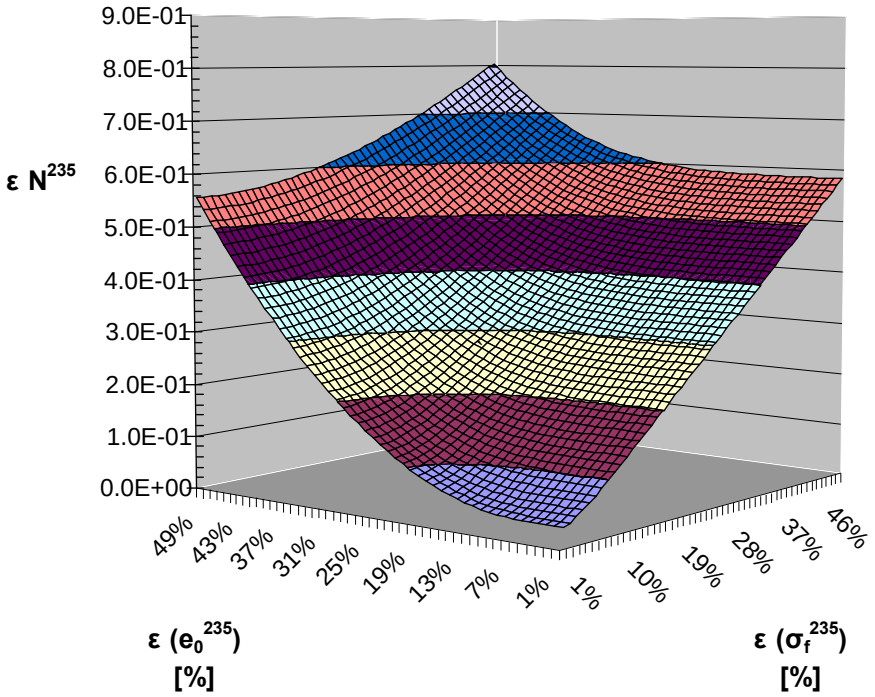


**Figure 2.** Error in the calculated $^{235}U/^{238}U$ value as a function of the error in the $^{235}U$ enrichment and the $^{235}U$ microscopic fission cross-section.

### 4.4. Sensitivity of error in calculated $^{236}U$ attribute

The sensitivity of the error in the $^{236}U/^{238}U$ value was determined by plotting the error in the calculated $^{236}U/^{238}U$ value as a function of the errors in the $^{236}U$ and the $^{235}U$ microscopic absorption cross-sections. The plot depicted in Fig. 3 shows that the calculated error in the $^{236}U/^{238}U$ value varies linearly as a function of the errors in the $^{236}U$ and the $^{235}U$ microscopic absorption cross-sections. The linear relationship determined indicates that the error in the $^{235}U$ microscopic absorption cross-section affects the overall error in the calculated the $^{236}U/^{238}U$ value more than the error in the $^{236}U$ microscopic absorption cross-section. This is because increasing the error in the $^{235}U$ microscopic absorption cross-section increases the overall error in the calculated the $^{236}U/^{238}U$ value significantly more than increasing the error in the $^{236}U$ microscopic absorption cross-section does. Therefore, more effort should be spent reducing the error in the $^{235}U$ microscopic absorption cross-section than reducing the error in the $^{236}U$ microscopic absorption cross-section.

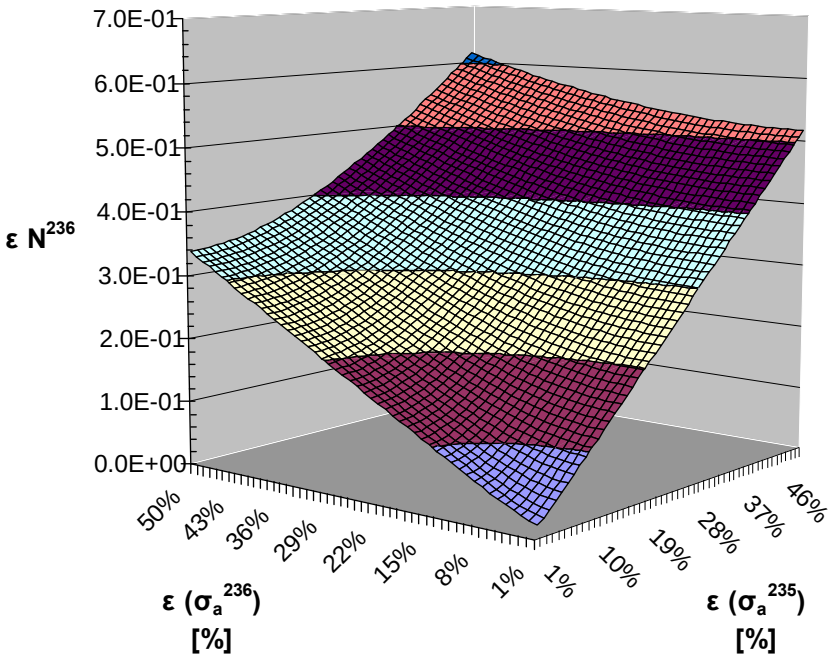


**Figure 3.** Error in the calculated $^{236}U/^{238}U$ value as a function of the errors in the $^{236}U$ and the $^{235}U$ microscopic absorption cross-sections.

## 5. $^{234}U$ Isotopics in Mines

After determining the enrichment process and the pre-detonation uranium isotopic ratios, this information may then be used to calculate the pre-enrichment $^{234}U/^{238}U$ isotopic ratio. Since different uranium mines throughout the world are characterized by different isotopic abundances of $^{234}U$, the source (mine or otherwise) from which feed uranium was taken can be determined by comparing the calculated pre-enrichment $^{234}U/^{238}U$ isotopic ratio to a set of measured $^{234}U/^{238}U$ ratios taken from various mining or milling facilities throughout the world. Thus, the natural variation of $^{234}U$ throughout the world provides a unique signature indicating the geographic origin of the material.

$^{234}U$ has a relatively short half-life and exists in secular equilibrium with $^{238}U$. Thus, the ratio of $^{234}U$ to $^{238}U$ should equal to the ratio of the half-lives (55.0 ppm). Variations in the ratio of $^{234}U/^{238}U$ may result from processes that disrupt the decay chain of $^{238}U$ to $^{234}U$ [5]. All of the measured $^{234}U/^{238}U$ values shown in Table 4 were determined using thermal ionization mass spectrometry where the $^{235}U+$ ion beam intensity was adjusted to correct for mass discrimination using the measured $^{235}U/^{238}U$ ratio obtained by gas source mass spectrometry.

| Sample No. | Country of Origin | Milling Facility | $^{234}U/^{238}U$ Atom Ratio | Statistical Uncertainty |
|---|---|---|---|---|
| 1 | Finland | Askola | 5.444E-05 | 8.0E-08 |
| 2 | Finland | Paukkajanvaara | 5.126E-05 | 7.6E-07 |
| 3 | Australia | Ranger Mine | 5.455E-05 | 4.4E-07 |
| 4 | Australia | Dam Operations | 5.341E-05 | 6.2E-07 |
| 5 | Canada | Cogema Resources | 5.385E-05 | 6.0E-07 |
| 6 | Canada | CAMECO Key Lake Op. | 5.397E-05 | 3.4E-07 |
| 7 | Gabon | Comuf Mounana | 5.434E-05 | 4.2E-07 |
| 8 | Czech Republic | DIAMO, Straz pod Ralskem | 8.355E-05 | 4.9E-07 |
| 9 | Canada | CAMECO Rabbit Lake Op. | 5.444E-05 | 4.8E-07 |
| 10 | Namibia | Roessing Uranium Mine | 5.460E-05 | 4.1E-07 |
| 11 | France | Cogema Lodeve | 5.154E-05 | 2.8E-07 |
| 12 | France | CETAMA Amethyste | 5.340E-05 | 3.3E-07 |

**Table 4:** Variations in measured $^{234}U/^{235}U$ atom ratios from mines throughout the world [5], [6].

A plot of the measured $^{234}U/^{235}U$ atom ratios with associated uncertainties for all twelve samples is depicted in Fig. 4. Sample 8 from the Czech Republic has a significantly greater $^{234}U/^{235}U$ atom ratio than any other sample which cannot be explained by geological processes. One possibility may be a result of anthropogenic contamination with plutonium, especially $^{238}Pu$ [5]. This contamination may have occurred as a result of the Chernobyl accident. A more in depth comparison of the variation in the measured $^{234}U/^{235}U$ atom ratios with associated uncertainties with sample 8 omitted is depicted in Fig. 5.

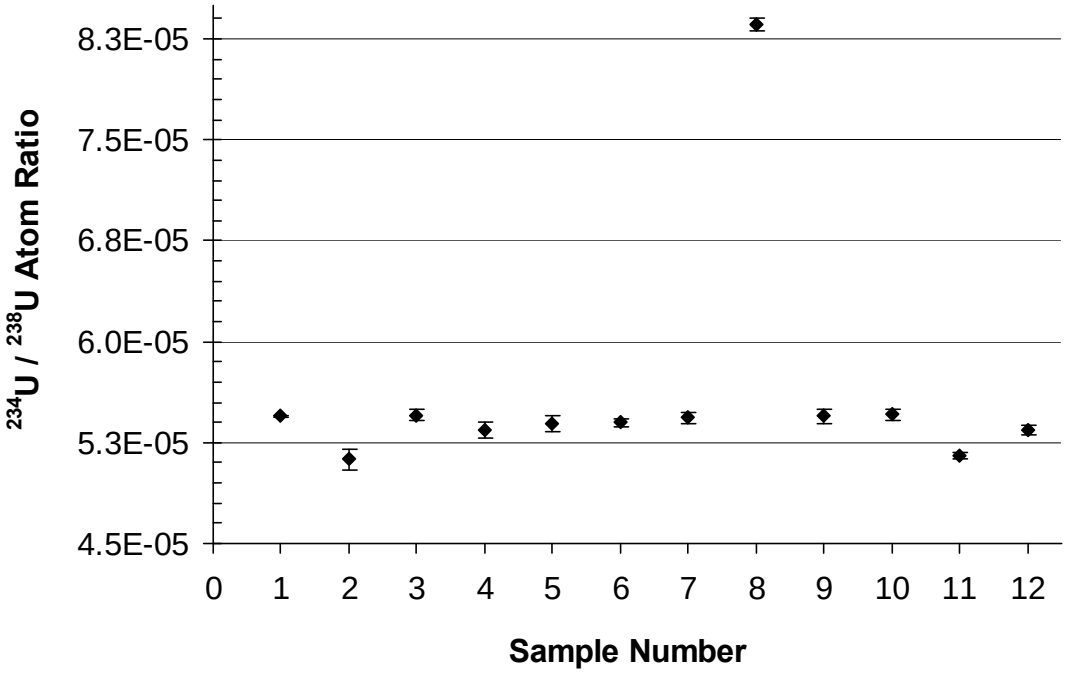


**Figure 4**. The $^{234}U/^{238}U$ atom ratio measured in all twelve samples.

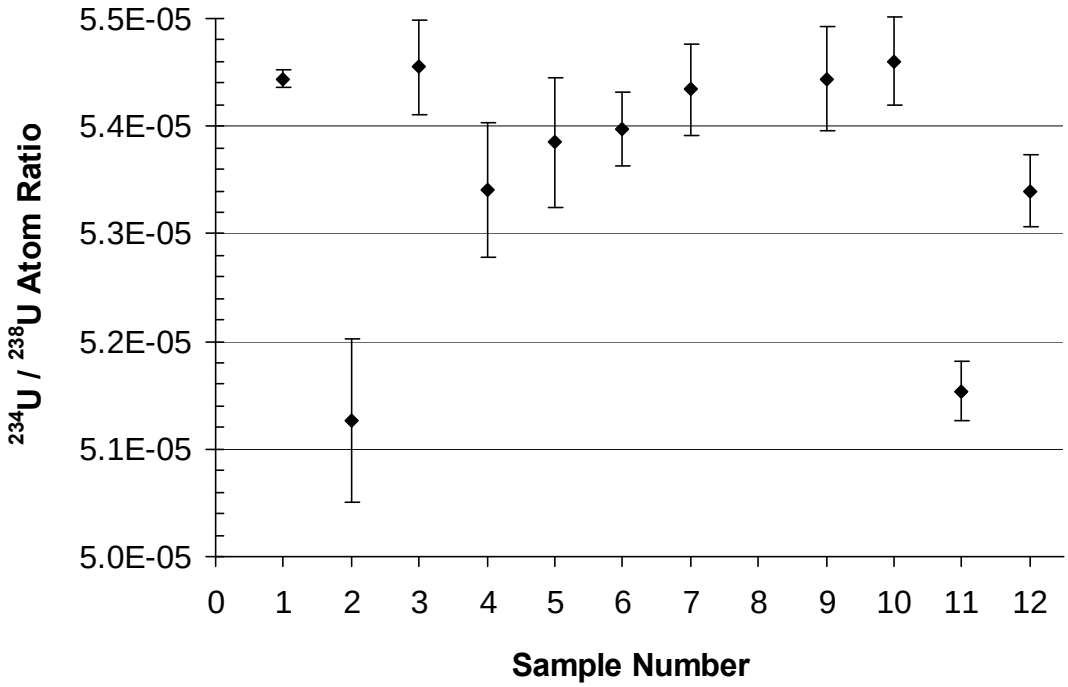


**Figure 5**. Expanded plot of the measured $^{234}U/^{238}U$ atom ratios excluding sample 8 [5], [6].

## 6. Discussion and Conclusion

### 6.1. Discussion

Various methods that could potentially be used to disguise the origin of the nuclear material used in HEU weapon prior to it being detonated were assessed in order to determine their effects on the validity of the algorithm. The first spoofing technique assessed was contamination of the original material used in the HEU weapon with fission products such as $^{137}Cs$ or $^{60}Co$. This will result in higher measured post-detonation concentrations of the fission products used to contaminate the original weapons material. The total number of fissions in the device per unit mass will be affected if the fission products used to contaminate the original material are the same as the fission products used in this calculation. Using two fission products the significantly increases the probability of determining that original material was contaminated because there is a smaller probability that the two fission products used in the algorithm were also used to contaminate the original material. If only one of the fission products that was used in the algorithm was also used to contaminate the original material, then the total number of fissions in the device calculated using one fission product will differ significantly from the value calculated using the other fission product. Thus, indicating that one of the fission products was either present in the original material or else measured incorrectly.

Another spoofing technique assessed was boosting the weapon prior to detonation. In a boosted nuclear weapon, a mixture of deuterium (D) and tritium (T) gas is injected into the central core of $^{235}U$ metal sphere, called the “pit”. The implosion of the pit causes the $^{235}U$ to fission which in turn causes the atoms in the D-T mixture to undergo fusion. The fusion reaction produces large quantities of high energy neutrons (approximately 14 MeV) which travel through the compressed pit causing additional fission reactions [7]. The boosting of a nuclear weapon greatly increases the yield by causing more of the material to fission during detonation. Therefore, if calculated yield of an HEU weapon was on the order of 100 kT or greater it was probably boosted. In the case where a weapon was boosted prior to being

detonated but was a fizzle, then the atoms in the D-T mixture did not undergo fusion and post-detonation measurements of both deuterium and tritium could be obtained.

The last spoofing technique assessed was using a combination of plutonium and uranium metal or Mixed Oxide fuel (MOX) fuel as the original material in the weapon. This presents the most difficult problem because not only will of the fission product concentrations be higher but any signatures indicating the method of enrichment will disappear. In this case, it might be useful to combine techniques used to determine the original material in both an HEU and plutonium device.

### 6.2. Conclusion

In this work, an algorithm was developed that uses measured isotopic ratios from fission products and actinides present following the detonation of a nuclear weapon to compute the original material attributes of the weapon. The algorithm was comprised of analytical inversions of first-order differential equations derived directly from burnup and radioactive decay equations. The following post-detonation isotopic ratios were used: $^{89}Sr/^{238}U$, $^{95}Zr/^{238}U$, $^{232}U/^{238}U$, $^{234}U/^{238}U$, $^{235}U/^{238}U$, and $^{236}U/^{238}U$. The primary advantage gained from this methodology was it provided accurate solutions with essentially no computational time required. Error propagations were used to determine the sensitivity of the error in the calculated original $^{234}U$, $^{235}U$, and $^{236}U$ attributes for the HEU fuel. The errors in the calculated $^{234}U/^{238}U$ and $^{236}U/^{238}U$ attributes were linearly related the errors in measured parameters. The error in the calculated $^{235}U/^{238}U$ attribute varied nonlinearly as a function of the $^{235}U$ enrichment placing a significant importance on ensuring the accuracy of this value. The determined signature that indicated the enrichment process used to create the weapons material was based on the measured $^{234}U/^{238}U$ ratio. A source of error that was not assessed exists in the cross-section data used throughout the algorithm from the ORIGEN2 library for an FFTFC reactor. In this work, we were only testing the feasibility of the algorithm and did not consider its relationship to an actual weapon detonation. Thus, testing of this methodology using cross-section data obtained for an actual device detonation would improve the viability of the algorithm.

This work is important to homeland security and a significant prototype to data protocol in the event of a terrorist attack in our country. The algorithm developed was restricted only to HEU devices; however, future efforts will consider plutonium devices as well. It is also necessary to analyze how elements disperse in the environment and what current technology is available to measure isotopic fission fragments in the environment. All of the above aspects will affect the validity of the algorithm and if it could in fact be used if a terrorist device was detonated in the U.S.

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# Session 24

# Plants (back end of fuel cycle)

# Safeguards Techniques at Storage Facilities - Operators' Thoughts and Experiences

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**Abstract:**

*Due to the inaccessibility of spent fuel in dry storage facilities an unattended system of safeguards instrumentation is needed to maintain continuity of knowledge on spent fuel contained in shielded flasks. Innovative C/S and NDA techniques, in particular with remote monitoring capabilities are appropriate to meet the safeguards requirements for such type of facility. However, before introducing these techniques into practice an operators' feasibility assessment is needed to evaluate carefully in advance the new measures taking into account operational and licensing boundary conditions of that particular facility. There are some basic operators' requirements which have to be considered also in view of the implementation of safeguards measures. Among these are radiation protection, safety at work and technical safety of the instruments, but also such features as reliability and non-intrusiveness are important.*

*For example metal seals, applied to spent fuel flask lids, have major deficiencies regarding radiation exposure of verification personnel during seal attachment/removal operations. Therefore, the preferred solution in this context would be the use of electronic seals which provide remote data interrogation capability. Another technique, which is presently under discussion, is remote monitoring, i. e. the transmission of status signals, measurement data and video images directly to the headquarters of EURATOM/IAEA. This technique could certainly contribute to the "non-intrusiveness" of safeguards measures. However, one of the major concerns is reliability of the system. Some basic questions concerning data integrity and security would have also to be answered to meet internal security requirements of the facility.*

*This paper gives an overview on safeguards technologies which are applied or under discussion for implementation at intermediate on-site storage facilities.*

## 1. Introduction

The Federal election in September 1998 has been a turning point with regards to the utilization of nuclear energy in Germany. The new governmental policy has been aimed at phasing out of nuclear energy and many aspects related to the generation of nuclear energy were put into question. Among such decisions as termination of reprocessing the philosophy of intermediate spent fuel storage until direct disposal has also been changed. As a result of comprehensive governmental discussions it has been agreed with the German nuclear industry that spent fuel will no longer brought to the away from reactor facilities at Gorleben or Ahaus. Instead, all spent fuel generated in the course of reactor operation will have to be stored in intermediate storage facilities at the reactor site. In total 12 new on-site storage facilities have been constructed to fulfil the governmental requirement. All on-site storage

facilities except one are presently in operation. The one remaining facility at Unterweser will receive the first flasks end of May 2007. During the last years more than 150 spent fuel flasks have been placed into the on-site dry storage facilities. Furthermore, until the end of 2009 another 140 flasks are planned to be stored. This development brought up the issue of spent fuel storage again and showed that there is an urgent need to focus on safeguards solutions which could cope with the considerable flow of spent fuel into the dry storage facilities and its intermediate storage for a period up to 40 years.

## 2. Facility features

Generally all German on-site storage facilities have a similar layout as the away from reactor facilities at Gorleben and Ahaus. They consist mainly of a reception, maintenance and storage area. However, there are two technical concepts which differ in such technical features as for example the kind of air conditioning and the number of storage halls. Furthermore, these two concepts are also different with regard to the kind of flask storage which may have an impact on the application of safeguards measures. Whereas the single-hall layout is characterised by a compact flask storage the two-hall concept is based on a storage of spent fuel flasks in groups with corridors between the flask rows (see figure 1). Figure 2 shows a photograph of the on-site interim dry storage facility at Lingen. This is the facility where the storage license has been granted first and it came into operation in late 2002.

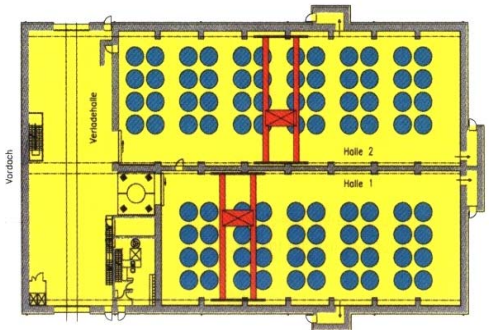


Figure 1: Floor Plan of an On-Site Storage Facility

It should be mentioned that there is a close connection between the storage and reactor facilities when considering safeguards measures for spent fuel storage. During the remaining reactor lifetime constant spent fuel loading activities and transfers to the storage site will be carried out. In view of the 12 on-site facilities this requires a rather big inspection effort since a final verification of spent fuel assemblies is needed before loading them into the flasks. Presently it is being considered how to proceed with these transfer activities and which measures could be applied to reduce the presence of inspectors during this period. Respective discussions on technical measures involve also the reactor operators. The paper does not emphasise on this particular problem but will rather focus on safeguards measures of the storage facilities.

Figure 2: Lingen On-Site Storage Facility

## 3. Preparation for Safeguards measures

When considering the safeguards approach for storage facilities specific aspects have to be taken into consideration. First of all, once the spent fuel has been loaded into flasks the inventory is no more accessible and respectively cannot be verified directly. Therefore, the safeguards measures applied to the storage facilities should be capable to maintain the continuity of knowledge on the flask inventory. This requirement is covered by using C/S measures and NDA techniques. Seals attached to the flasks and TV surveillance of the areas where the flasks are handled and stored supported by NDA techniques play a major role with this regard and generally form the basis for a safeguards approach. However, the devil is in the detail and different boundary conditions have to be taken into account. A close cooperation between the operators and the inspectorates is inevitable to find a safeguards solution which is acceptable for both the inspectors and the operators. Of course, the operators are obliged and willing to fulfil the safeguards rules but they have also to comply with the national regulations related to safety at work, radiation protection, technical safety of instruments and physical protection. Consequently, any measures, including safeguards instruments, applied to the facility must not be in collision with the general and plant specific rules.

In view of the construction and operation of the 12 new storage facilities an exchange of information between the facility operators and EURATOM inspectors has been started early. The operators were in particular interested to know which technical infrastructure is needed and what the specifications of the technical means are to realise the planned safeguards approach. These technical needs were defined by the inspectorates already at the end of 2001 and were taken into account during the planning and construction of the storage facilities. At that time the safeguards approach was not yet defined finally. Therefore, an extended approach formed the basis for the application of safeguards measures at on-site storage facilities which included video surveillance in the reception, maintenance and storage areas, neutron monitors in the reception/maintenance area and different kind of seals at the flasks (see figure 3). To realise this instrumentation concept various constructional and technical measures had to be prepared by the operators as for example:

- Anchor plates at the positions for video cameras and detectors
- Cable installation into the cable ducts
- Wall openings for the cables
- Space for the surveillance system cabinet
- Emergency power supply for safeguards instruments and lightening
- Separate ISDN-line for remote data transmission

At present the preconditions for the installation of the safeguards techniques are realised at all on-site storage facilities.

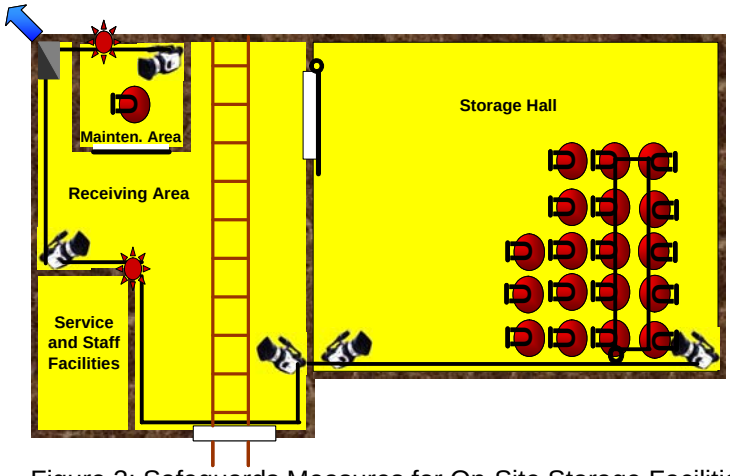


Figure 3: Safeguards Measures for On-Site Storage Facilities

## 4. Seals

The application of seals to the flasks is an effective safeguards measure from the inspector's point of view but it creates difficulties for the operator in general and in particular. Generally, it has to be stated that the technical concept of a storage facility is such that a regular visit of the storage hall is not necessary and should be avoided for radiation protection reasons. The operator will enter this area on

a need-to-do-basis only. By this facility concept the ALARA principle is followed, i.e. the radiation exposure of the personal is kept as low as reasonably achievable. Seals on flasks violate this principle. In particular, when metal cap seals have to be verified in close proximity to the flask, i. e by removing them and attaching new ones the radiation exposure to the inspectors and the accompanying personal is unacceptably high. Therefore, this kind of seal cannot be a solution for an on-site storage facility which will steadily increase its number of flasks up to a maximum of 192 in the largest storage facility. Unfortunately, these old-timer seals are still very popular and used for flask sealing despite their deficiencies.

To reduce radiation exposure different types of seals should be used providing for a short-term presence of inspectors and operators in the storage hall for interrogation purposes. The COBRA fibre optical seals are now regularly applied to spent fuel flasks in a way that the sealing body is at the inspectors' eye level so that the short interrogation is possible without climbing to the top of the flask. Through a special screw cap the unauthorized removal of the screw and the seal is excluded. The COBRA seals constitute definitely a progress in comparison to metallic seals but still require a close contact to the flasks. In on-site facilities with compact flask storage the inspector even has to slip into the narrow space between the flasks where he is not only exposed to radiation but to high temperatures as well. Here, a better solution would be the use of seals with remote interrogation capability like the new generation of the electronic seal type EOSS. This kind of seal is equipped with interfaces allowing seal interrogation remotely from the outside of the storage hall. EOSS seals are planned to be used for the sealing of a group of flasks. Although the attachment of the group seals needs also a close contact to the flasks the advantage of this sealing mode is paramount.

Apart from the radiation aspect there is also the aspect of safety at work. Seal operations like group sealing or metal cap seal exchange, when necessary, have to be done at a flask height of 6 meter. Specific auxiliary equipment is therefore needed to ensure the safety of inspectors at such heights. Difficulties arise in this context when the flask storage is very compact without corridors between the flask rows as in the single-hall layout.

The operators view on flask sealing can be summarized as follows: They compromise on the application of COBRA and electronic seals with the aim to reduce the on-site interrogation activities in case of COBRA seals to the absolute minimum and focus on the remote interrogation of electronic seals from outside the storage hall. The use of metallic cap seal is highly problematic as mentioned above. From the operators view this seal could play a role as back-up measure only in case the other seals fail. In this context the IAEA rule of a restricted application period of three years is not helpful and an exemption from this rule should be agreed upon.

## 5. Video surveillance

The second important C/S measure is the surveillance system. Digital image surveillance systems are presently state of the art for safeguards. Video surveillance requirements in the storage hall would be covered very well by the multi camera system of the SDIS-type which has reached a high technical standard. Generally, there are no major problems identified to utilize optical surveillance in storage facilities. Of course, the operator has to take special technical measures to ensure an undisturbed function of the surveillance system. In particular, these measures concern specific circuits for a permanent lightening of the monitored areas and an uninterrupted power supply for the surveillance system cabinet. On the other hand the operators have to comply with German technical standards as for example the VDE and DIN norm for the installation of any technical equipment in the facility. The safeguards instrumentation provided by the inspectorates to be used in the storage facility has to be conform to these norms and rules as well, i.e. these instruments should have a professional standard. This is the case with the SDIS system mentioned above.

## 6. Remote data transmission

One aspect of the implementation of the electronic instrumentation mentioned above is its potential for remote data transmission (RDT). Also the type of facility which is characterised by only a few movements per year is predestined for the application of RDT. Remote monitoring of safeguards data could considerably contribute to the optimisation of inspection activities with the aim to reduce the

inspection effort. The latter would also positively influence the operator's effort by reducing the escorting activities and the radiation exposure simultaneously.

The operators are generally open to the introduction of RDT. However, there are a number of general and specific questions which have to be solved before. For that reason discussions with the inspectorates and German competent Ministries were held recently.

There is a common understanding of the operators that the principle of periodic safeguards activities at the facilities should remain valid. The RDT represents a technical means and should serve for optimisation purposes only. A changing to permanent facility monitoring by permanent data transmission and frequent evaluation of the data is not compatible with the current safeguard practice. Furthermore, from the operator's point of view the data transmission must be restricted to safeguards data only. Any transmission of operational data with no safeguards relevance must be excluded.

The relevant safeguards data to be transmitted could be state of the health data of the equipment, measurement data, video images and data on the seal status. The boundary conditions for each kind of data to be transmitted have to be analysed and defined separately. Concerning the transmission of digital surveillance images the operator claims the right for a delayed transmission of these data (for example 24 hours). Thereby the operator would be in a position to evaluate the information and to react immediately before data transmission has occurred. On the other hand, the real-time transmission of state of the health data is acceptable and has already been carried out in some nuclear facilities.

The safety and security of the plant has the first priority and should not be violated by the remote transmission of safeguards data. The main principles of the plant IT-system need to be fulfilled also in context with the transmission of safeguards data. In particular, data security has to be ensured and loss of confidentiality, integrity, availability of the transmitted data must be excluded.

The inspectorates envisage to apply also remote system administration. This issue needs further negotiations since at the moment the boundary conditions for the implementation of the system administration are not yet defined sufficiently.

Finally, it should be mentioned that the system of remote data transmission is of no value if a reliable function of the system is not guaranteed. In case of failure all benefits and savings for the inspectors and the operators could lead to a number of additional measures which may become necessary to overcome anomalies caused by the technical failure.

## 7. Present Safeguards practice

All on-site storage facilities except one are in operation and the present number of flasks stored in these facilities varies between 3 and 36 flasks. The operating on-site facilities are inspected regularly according to the defined inspection regime. Safeguards measures like seals and video surveillance are applied to all operating facilities and ensure the continuity of knowledge. In contrast to the extended safeguards approach mentioned above no neutron monitors will be used in on-site facilities and the number of cameras will be reduced and installed in the maintenance and storage hall only. The realization of safeguards measures according to the agreed concept differ from facility to facility. In a number of plants the instrumentation is still provisional and the infrastructure prepared by the operators is utilized only in a few facilities. Mostly a single ALIS camera is provisionally installed at the facilities for surveillance purposes. Furthermore, in many cases two metallic seals are attached to the flasks, one at the secondary and protection lid each. In addition the COBRA seal is attached to the protection lid serving for the regular re-verification of the flask integrity. Group sealing with electronic seals has not been implemented yet in any on-site storage facility although there are some on-site storage facilities where the flask number is already considerably high.

## 8. Conclusion

The planning of 12 on-site storage facilities with a capacity between 80 and 192 flask positions has led to early communications between the operators and the inspectorates aiming at defining the

safeguards measures to be implemented and the respective constructional and technical infrastructure to be prepared by the operators. While the latter is realized in all on-site facilities the planned instrumentation concept by the inspectorates is not yet implemented completely. There is still potential for improvement but the operators are optimistic that in the near future all the envisaged safeguards measures, including the group sealing will be implemented at the on-site storage facilities.

# The management of spent nuclear fuel and radioactive waste in Sweden.

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## BACKGROUND

The foundation of the policy and distribution of the responsibilities for the radioactive waste management and its final solution was done already in the 1970's. The "Stipulation Law" (1976) states that the producer must take full responsibility of the waste generated during the power production.

SKB (earlier SKBF) was established 1972 by the Swedish nuclear utilities with the mission to manage the backend cycle for radioactive waste and spent nuclear fuel. During the 1980's the "Finance Act" was established with the objective to allocate funds for the management of the complete backend cycle for radioactive waste and spent fuel.

Looking back from the situation today with the nuclear waste management programme, a lot have changed since nuclear power was introduced in Sweden.

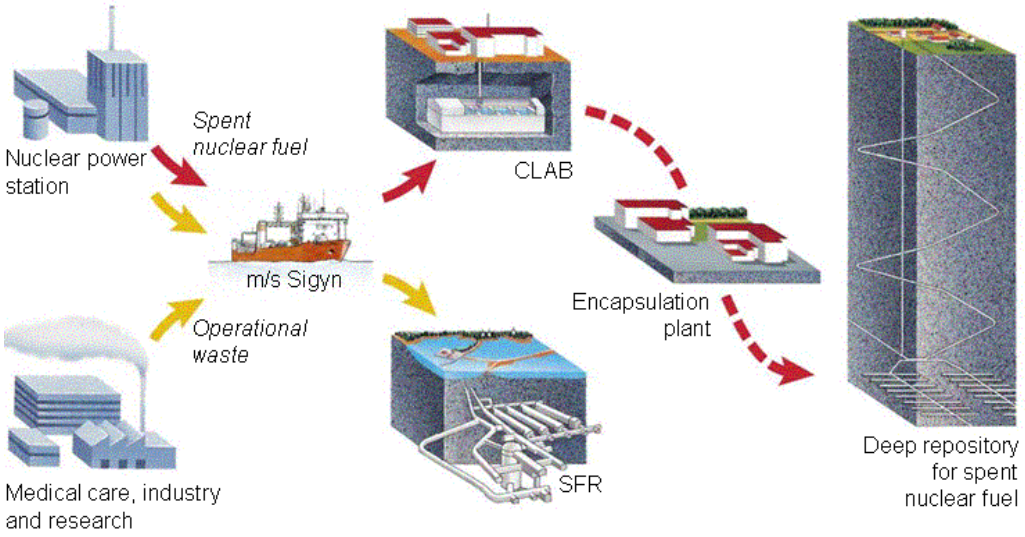
- Reprocessing is no longer part of the strategy and only a minor amount of fuel from the first reactor will ever be reprocessed. Instead a central storage facility for spent nuclear fuel, CLAB, has been built and been in operation since 1985. Today about 4700 tonnes are stored there.

- A system for transportation of spent nuclear fuel and low- and intermediate level waste from the reactors is in operation since 1985.

- A final underground repository for low- and intermediate level waste, SFR, is in operation since 1988. Thus a long-term solution has been established for this type of wastes.

- The work on research, development and demonstration for deep geological disposal of spent fuel has been intensive for more than 20 years. It has now reached such maturity that key decisions concerning design and siting of an encapsulation plant and a final repository are possible to be taken within the next 5-10 years.

- In November 2006 the application for the encapsulation plant was submitted to the authorities



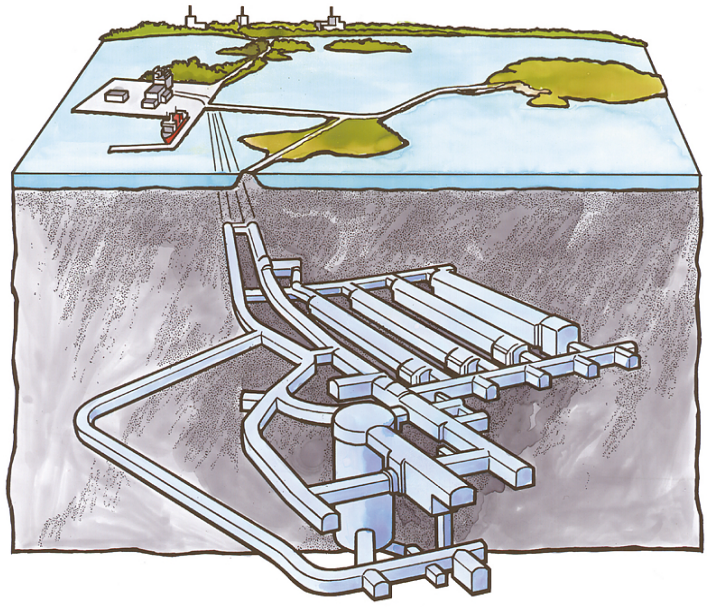

*The Swedish system for waste management*

## FACILITIES IN OPERATION

### Final Repository for Low and Intermediate Level Operational Waste, SFR

SFR, the Swedish Final Repository for Radioactive Waste, is designed for short lived LLW/ILW from the operation of all Swedish Nuclear Power Plants. The repository also accepts waste from research, medicine and industry providing the waste have similar properties and radionuclide contents as the waste from the power plants.

The repository has been in operation since 1988 and has a capacity of 63 000 $m^3$ waste. The license for SFR allows 90 000 $m^3$ waste. Until the end of 2006 SFR has received 31 249 $m^3$ waste.
The repository is located close to the nuclear power plant at Forsmark, in crystalline bedrock, 60 m under the bottom of the Baltic Sea. SFR consists of an above ground section and an underground section. The above ground section consists of office, workshop, terminal building for transport containers and the ventilation building.



SFR Repository

## Experience of Operation

*Transport*

The waste packages are transported to SFR with a specially designed ship, the M/S Sigyn. It was built in 1982 specifically for transporting spent nuclear fuel and radioactive waste. The ship has a roll-on/roll-off design, and carries up to ten transport containers at a time.

Even though the ship was purpose-built, safety is primarily guaranteed by the transport containers, which provide radiation shielding and mechanical protection in the event of an accident. The containers meet the IAEA Type A requirements. These containers weigh up to 120 tonnes when filled with waste packages. To move the container to and from the ship, specially built transport vehicles are used.
LLW that does not require radiation shielding during transport and can be transported in standard freight containers (ISO containers).

*Waste Packages*

Since there are four power utilities in Sweden and in addition a research centre producing waste packages for disposal in SFR, there has to be some co-ordination in the design on the packages. For that reason most of the packages disposed of in SFR have a standardised geometry.

The intermediate waste especially filters and ion exchange resins are mixed with cement or bitumen and cast in cement or steel boxes or metal drums. Some filter and ion exchange resins with lower activity content are packed in concrete tanks and dewatered.
Also some packages with non standardised geometry have been disposed of in SFR.
For example a PWR reactor tank lid from Ringhals power plant has been disposed of in one of the rock caverns.

*Operation of the facility and monitoring studies*

The SFR repository has been in operation for 18 years. The technical systems and the facility have been functioning very well.

The receiving capacity at SFR is 6000 $m^3$/y. The full capacity implies two shift works which have never been necessary to use. During the first years of operation the annual amount of waste was 2000-3000 $m^3$. Now, when the interim storages at the power plants have been emptied it is only the new production that is disposed of. This corresponds to approximately 1000-1500 $m^3$/y from all NPP.

## The Interim Storage of Spent Nuclear Fuel, CLAB

The Interim Storage Facility for Spent Nuclear Fuel, CLAB, was commissioned in 1985 and the first transport of spent fuel was received in July the same year. The decision to build CLAB was taken 1979 but the construction start was delayed until late 1980 because of the Swedish referendum about nuclear power that was held in March 1980. CLAB is located at the peninsula of Simpevarp close to the Oskarshamn nuclear power plants and is the interim storage for all spent nuclear fuel from the Swedish nuclear power plants.

The facility is a wet storage with its storage pools in a rock cavern about 30 meters below ground level. The original capacity was 3000 tonnes of uranium but by changing the design of the storage canisters the capacity was increased to 5000 tonnes. The original canister contains 16 BWR or 5 PWR fuel assemblies and the new “high density packing” canisters contain 25 or 9 fuel assemblies, respectively. The high density packing as enabled by using boron alloyed stainless steel in the compartments for the fuel assemblies.

In 1998 a decision was taken to extend the storage capacity to 8000 tonnes of uranium/HM by building another rock cavern with storage pools parallel to the existing one. These new storage pools is planned to be taken into operation late 2007.

*The Central Intermediate Storage for Spent Fuel (CLAB)*

The transport of spent fuel is performed by M/S Sigyn, a specially designed ship for transport of radioactive waste and spent fuel.

The transport cask used for the transport of spent nuclear fuel is TN-17 Mk 2, which is a cask for dry transports. This means that the cask and the fuel must be cooled before unloading into the pools. This is performed by an external cooling circuit for the cask body, and an internal circuit for the fuel itself.

Until December 2006, CLAB has in total received more than 23000 spent fuel assemblies corresponding to 4774 tonnes of HM (initial). Spent fuel from the PHWR prototype reactor Ågesta and some MOX fuel from German nuclear power plants are also stored at CLAB. Highly radioactive material such as control rods and components from internal parts of reactors are stored in the pools.
In total have about 1600 transport casks been received and handled. M/S Sigyn has travelled more than 520000 nautical miles for transport of radioactive waste.

The experience of the operation including the handling of the transport casks is very good and demonstrates that the CLAB facility is a safe, reliable and robust facility.

## FUTURE PLANS

The safety in the long term perspective is based on the isolation function of the repository. The technical solutions that have been studied are based on the following principles:

- Final disposal in Swedish crystalline bedrock
- Independent natural and technical barriers, the multi-barrier principle.
- Natural materials in the technical barriers.

- Limited temperature, radiation and other impact on the host rock

Based on these principles the details of the Swedish KBS-3 method have successively developed into the larger context of a total system. The key issues have e.g. been:

- Spent fuel encapsulation technology and canister design
- Design, manufacturing and practical emplacement of the bentonite buffer
- The adoption of repository design to the properties of the host rock
- Scientific basis for understanding of the long time functions, e.g. concerning corrosion of the canister, dissolution of the fuel and the migration of radionuclides in the buffer and in rock fractures.

Today SKB is operating the Äspö hard rock laboratory for research and demonstration of the disposal technology.

## Encapsulation of Spent Fuel

The spent fuel will be encapsulated in a canister of solid copper. This canister itself is the safety concept and the design criteria are that the copper canister isolates the spent fuel from the environment for at least 100000 year. Copper is chosen because of its excellent corrosion properties in the environment of Swedish granite at a depth of about 500m. Inside the copper canister is a cast iron insert to fulfil the strength requirements associated with withstanding the hydrostatic pressure, in worst cases during a glacial period.


**50 mm copper**

**Estimated weight (kg):**

| | |
|---|---|
| Copper canister | 7,400 |
| Insert | 13,600 |
| Fuel assemblies (BWR) | 3,600 |
| Total | 24,600 |

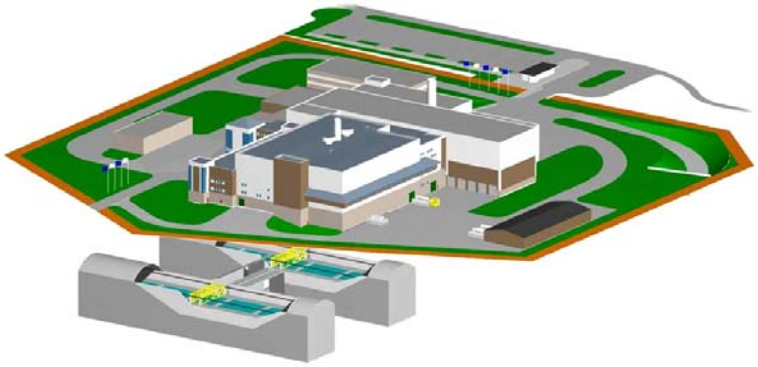
*Copper canister for spent nuclear fuel*

The encapsulation plant can be sited adjacent to the central interim storage, CLAB, the deep repository, an existing nuclear facility or somewhere else. The current plan is that the Encapsulation Plant will be as an extension to CLAB. This location provides possibilities to

extend several existing service systems into the Encapsulation Plant and share personnel and other functions. The Encapsulation Plant consists mainly of two buildings, an encapsulation building and a storage building for transport casks. The encapsulation building is connected to the CLAB underground storage pools via the existing fuel elevator.

The main parts of the controlled area in the encapsulation building are a pool section, a hot cell section and a dispatch hall where canisters are prepared and imported/exported. In the non-controlled areas the power supply systems and non-radioactive auxiliary systems are installed.

The plant will be approximately 65x80 in size and about 25 meters high, which is equivalent to the height of the existing receiving building at CLAB. It is designed for an annual output of approximately 200 Disposal Canisters per year, i.e. on the average one canister per workday. The operating staff will be shared between the Encapsulation Plant and in CLAB.



*The encapsulation plant (in front) located in connection to the CLAB facility*

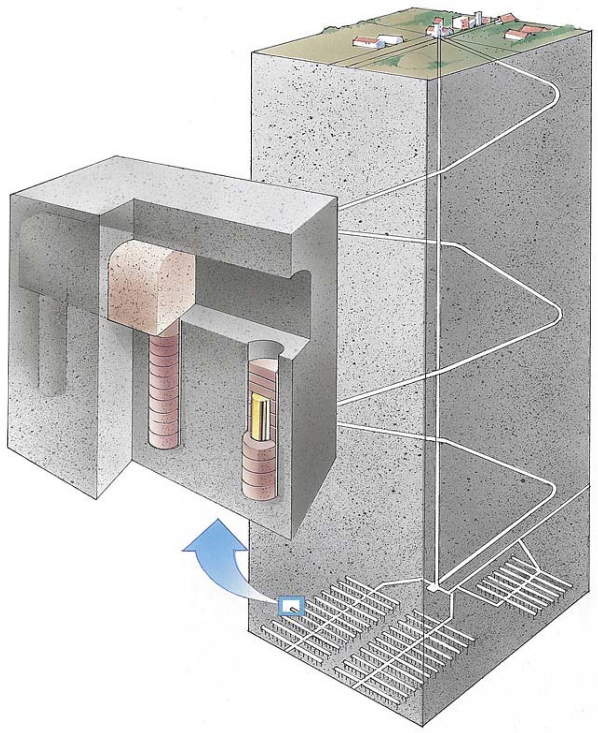
The fuel assemblies stored in CLAB have very different burn-up and residual power. These properties determine the heat output of the canisters, which is a restricting factor in the geological repository. To minimize the total number of disposal canisters, the combination of fuel assemblies in a canister has to be optimized. Therefore, and also for safeguards, the fuel assemblies can be measured and sorted in a pool before encapsulation.

The canister will be welded together by special welding equipment. When the weld is completed, the disposal canister is transferred for non-destructive testing and machining. The technologies for both welding and testing are currently being developed and demonstrated in full-scale at the Canister Laboratory in the town of Oskarshamn. Several technical alternatives are tested in parallel, such as electron beam and friction-stir welding and ultrasonic, x-ray and eddy current inspection techniques. The friction-stir welding has been chosen as the reference method.

## Disposal of Spent Fuel

The waste package (the copper canister with the spent fuel) will be transferred from the Encapsulation Plant in a transport cask to the repository. The geological repository consists of a surface structure with buildings for receiving and service for the underground activities. At about 500 m depth the waste package will be disposed in either vertical or horizontal holes in drifts. In the deposition hole the waste package will be surrounded by bentonite. When the

deposition holes are filled in a drift the drift will be backfilled with a mixture of granite (from the excavation) and bentonite.



*The Geological Repository for Spent Nuclear Fuel*

**Handling of Spent Core Components**

Apart from the spent fuel, another important long-lived waste category is the core components that have been replaced when reactor internals have been upgraded or repaired. Although the CLAB facility is designed to accept spent core component, these scrapped components are often, for practical reasons, stored on-site at the nuclear power plants. During recent years, methods have been demonstrated to segment these components into manageable pieces and packed in volume-efficient boxes.

**Disposal of Spent Core Components**

A special repository for metallic core components and internal parts with a high specific activity from nuclear reactors will be designed and built. The facility is called the repository for long-lived low- and intermediate level waste. It will, according to the current plans, be an extension of the existing repository for low- and intermediate level operational waste, SFR, but with the disposal rock caverns positioned much deeper into the bedrock, at a level of about 300 meter below ground.

## Safeguards aspects

*The facilities*

As operator one MBA is preferable for the interim storage Clab and the encapsulation plant in order to keep the administration to a minimum. This will also be good for the authorities.
The repository will be one MBA
In the encapsulation plant first item is the fuel assembly and after encapsulation the copper canister will be one item.

In the encapsulation plant there will possibilities to perform measurements on the fuel assemblies and there is also a special room that is intended for safeguards equipment and dedicated for use by authorities as Euratom, IAEA and SKI.

During the construction of the repository there will be opportunities to verify the tunnels and excavations of other underground spaces. This can be done by laser technique and also by determine the masses/volume excavated.
Monitoring equipment can be installed at the entrance of the tunnel to the repository so it can be verified that the transport cask with the copper canister is filled when it arrives and emptied when it leaves the tunnel entrance.

The disposal concept itself makes it very difficult to get access to the spent fuel without that this should be discovered.

*Challenges*

Today there are no instructions or recommendations from the regulatory bodies nor from Euratom and IAEA what a “safeguards system” at an encapsulation plant or a repository shall comprise.
At the encapsulation plant it will be the “last” time when the spent fuel is accessible and can be verified. This means that Continuity of Knowledge will be a very important tool for the safeguards system at the encapsulation plant and the repository.

If the verification of the fuel assembly will be on pin level that demands that all rods as specified in the fuel assembly is in place a lot of research and development of techniques must be performed. Today there are no methods that can do this in a practically and reliable way.

Integrated safeguards can be an important element in the verification of the facilities and activities.

It is very important to have a holistic approach to safeguards and don’t look to much on the facilities separately and instead look on the whole fuel cycle.

## Time schedule for future activities

The application for the repository is planned to be submitted 2009. If everything runs smoothly the encapsulation and disposal of the spent nuclear fuel can begin 2018.

# Database for Storage and Evaluation of Radiation Profiles taken of Dry Spent Fuel Storage Casks

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***Abstract:***

*IAEA safeguards are applied to dry storage of irradiated CANDU bundles in several countries. Before the casks are sealed, radiation traces are measured along the irradiated fuel stack to support verification of the loading and make provision for possible re-verification as necessary.*

*A database for the storage and evaluation of "fingerprints" (DSEF) has been developed to secure the measurement data over long periods of time and enable their easy retrieval to compare baseline radiation profiles with more recent measurements, thus supporting verifying the absence of retrieval of nuclear material from the casks.*

*DSEF is a distributed application allowing stand-alone operation in the field before synchronisation with data in the central database.*

*DSEF incorporates advanced evaluation features aimed at recognizing the number of fuel baskets or modules loaded in the casks and assessing the similarity of radiation profiles taken at different times.*

*To recognize the presence of modules and baskets DSEF implements physical models of the various CANDU storage designs (MACSTOR, DSC and Silo). Using the physical model of the propagation of gamma rays from the baskets to the detector, DSEF rebuilds a theoretical gamma or neutron emission pattern consistent with the experimental data. Then the number of baskets involved in the theoretical gamma emission pattern is counted and compared to the declared number of loaded baskets.*

*To fulfil the radiation profile comparison goal, DSEF algorithms correct the data for the radioactive decay, and the differences of data taking into account parameters like the motion speed of the probe, the efficiency of the detector or the dead time.*

*In addition to the data evaluation features and performances, the paper describes in detail the software architecture and its integration in the IAEA Safeguards IT system.*

**Keywords:** pattern matching; CANDU storage; radiation profile; monitoring

# 1. Introduction

The Department of Safeguards of the IAEA safeguards CANDU[1] spent fuel bundles stored into MACSTOR[2], silo or DSC[3]. Amongst other measures, the IAEA records radiation profiles on the CANDU dry storages to verify the initial loading of the casks and also as provisions for further analysis to restore continuity of knowledge as necessary.

MACSTOR and silos have verification tubes allowing vertical scans of the stored nuclear material to record gamma and neutron spectra. DSC have verification tubes allowing only gamma profiling.

The Division of Technical Support initiated a project to provide the Divisions of Operation with a software solution aiming at securing and evaluating the radiation profiles taken in the field by the IAEA inspectors

A contract was awarded to EURIWARE for the development of the Database for Storage and Evaluation of Fingerprints (DSEF). The Division of Technical support developed the feature requirements while the Division of Information Management gave guidelines for smooth integration in the IAEA information system.

The main purposes of the contract were:

- to design and implement a database to store neutron or gamma spectra generated by various kinds of equipments;
- to design and implement comparison algorithms supporting the similarity assessment of the measured data taken at different times;
- to design and implement methods to verify the number of baskets loaded in the dry storage casks from the analysis of the radiation traces.

The concepts underlying the development of DSEF were:

- to establish a unique tool for securing and processing the radiation profiles taken by the IAEA;
- to allow use of the database in standalone mode on inspectors' laptop while all data could be secured in a centralized database;
- to ensure maintainability of the application;
- to incorporate data evaluation features focusing on radiation traces taken on CANDU dry storage while allowing to store any other radiation traces.

This paper mainly focuses on the description of the methods and algorithms developed for processing the data.

# 2. Problem definition

## 2.1. Measurement accommodation

Different parameters may influence the spectrum for a given cask:

- the collimated probe can be adjusted differently between two measurement campaigns,
- the probe speed may change
- the measurement can be made either downward or upward,
- the probe or its electronic part may be changed between two campaigns,
- the dual time may be different and for a given probe speed the channel number is different,
- the probe may be partly blocked during the motion within measurement tube.

This measurement accommodation needs then to implement different kinds of algorithms such as:

[1] Canadian Deuterium Uranium Reactor
[2] Modular Air-Cooled Storage
[3] Dry Storage Container

- Merging data to produce signal versus location data by knowing the dwell time, the motion speed and scan direction or by merging radiation data file with location data file,
- Dead time correction by applying an average dead time correction or a point dead time correction when the dead time value is known for each measurement point,
- Smoothing by applying a moving average over a selected number of channels,
- Scan speed correction in case of positioning signal recorded with radiation traces, to assume that the probe motion is steady along the radiation trace,
- Cross calibration of detectors used for each measurements,
- Normalization if the cross calibration can't be performed.

## 2.2. Spectra comparison

The DSEF application has to compare a new measurement with a baseline measurement taken at the end of all nuclear materials movements. The new measurement and the comparison report must be stored in the database for further utilization.

The comparison result must be trusted and must answer to the question: am I sure that no change occurred and can I be sure that a change occurred. For instance the answer may be the result of a comparison between a spectrum on 300 channels and one on 6000 channels. Ageing diode whose efficiency decrease, different initial stroke between spectra are other parameters that disturb signal comparison. Moreover, we may have to compare a baseline acquired downward with an upward measurement acquisition.

This comparison can be performed by applying different kinds of algorithm such as:

- Offset correction done by using signal inter-correlation. In this case, the offset value is determined by the maximum of the inter-correlation function between the baseline signal and the measurement signal.
- Decay correction according to $^{137}Cs$ decay for gamma and $^{244}Cm$ for neutron.
- Similarity between the baseline and the measurement can be given by the use of distance algorithms. The comparison between the result of the selected algorithm and a defined threshold allows determining if the two signals are similar or not. A level of confidence must be implemented for each diagnostic
- Automatically count of the number of baskets in a CANDU silo. This count can be done by using the gamma radiation traces on which each basket produce a local maximum.

The most difficult part of the DSEF implementation is the comparison between two spectra.

The main problem is to quantify the deviations observed along the spectra. If a threshold must be used, its value must take into account the background noise of the two spectra. Furthermore, small local deviations are more significant than larger deviations distributed all along the spectrum.

## 2.3. Basket counting

Signal interpretation to count baskets reveals itself very difficult. First of all it is difficult to detect the upper basket due to measurement tube bend. Other phenomenon is dependant on heterogeneous activity distribution between baskets which can hide a transition between two baskets. Finally the concrete rebar disturbs the signal (addition of a noise with high magnitude).

Attempt to detect baskets on plots was sometimes impossible even for trained eyes.

## 2.4. Problem redefinition

Analyzing different kinds of spectra leads us to a new way of treatment for silo and MACSTOR. For these two types of storage, spent fuels are stored in vertical baskets. However the theoretical number of basket is known. The idea is to first locate these baskets in the spectrum, and secondly to work on each basket characteristics, instead of performing spectrum analysis first. A comparison between two measurements will then be reduced to a comparison between the areas of found baskets.

The problem becomes therefore to find a defined shape along a spectrum. Then for silo and MACSTOR the problem becomes a model identification problem with a spectrum fitting procedure.

The questions then become: What is a basket? How can we characterize it in such a way that we can manage geometry into spectrum shape? A modeling effort was then necessary to validate the approach before starting development.

This modeling effort consisted in:

- Describing the geometry in a few parameters
- Validating this model with fine simulation
- Finding a “correlator” able to distinguish differences which are inexplicable (lack of basket, removed material) from those which are not (in case of collimation device problem).

However, for the DSC storages, the fuel bundles are not stored in vertically stacked baskets, so, it appears that only the comparison between the spectra is possible since there is no invariant that can help.

## 3. Modeling for silo and MACSTOR

### 3.1. Basket shape definition

We look for a model of the shape that allows to find in the spectrum.

Signal shape due to a basket can be modeled by a formula depending on:

- the active length of basket,
- the distance between the basket axis and the measurement tube,
- the total activity of the basket.

We need to model the activity measured by a probe moving on a vertical axis parallel to a basket containing radioactive material in bundles (fig 2). The axis of verification tube is in the concrete of the silo and is surrounded by measurement tube as show in figure 1.

The following simplified problem can be studied:

- the basket is constituted of radioactive material distributed in homogeneous manner in a cylinder
- the space is completely transparent to radiations. We don't take into account the attenuation due to concrete and metallic structures.

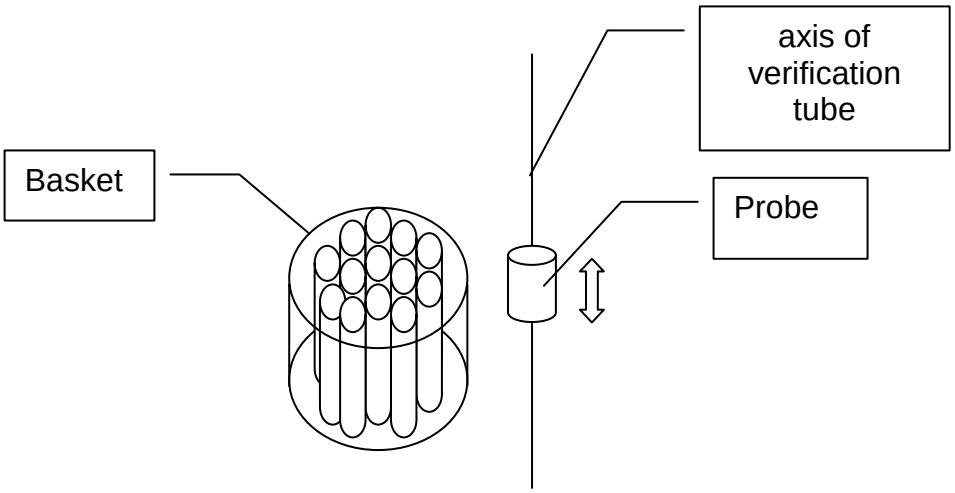


**Figure 2:** Geometrical model of a basket with a probe located in front of it

Due to the symmetry, and in applying the gauss theorem on a cylindrical surface concentric to basket, the measured activity (as it is seen from the probe) is the same as if all the radioactive material was concentrated in an infinitely thin segment centered on the basket axis and with a same length as the basket one (fig 3).

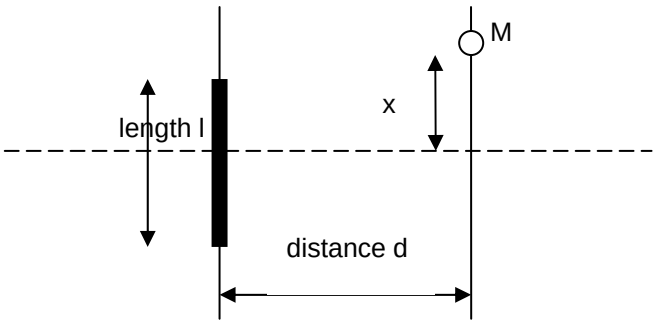


**Figure 3**: Equivalent geometry definition

### 3.2. Activity definition

Integration of radioactive segment points at the position of M gives the activity at M:

Activity at M = A * [ atan((-l/2 - x)/d) – atan((l/2 - x)/d) ] / (l * d),

where A is the total segment activity.

After normalization at the point x = 0 the formula becomes:

Activity at M = [ atan((-l/2 - x)/d) – atan((l/2 - x)/d) ] / [ atan((-l/2)/d) – atan((l/2)/d) ]

An accurate simulation was made with MCNP[4] on current geometry data.

The result is shown in figure 4. Curves are fitting for *l* = 52,66 cm and *d* = 2,109 m.

The *l* and *d* values are determined in order to fit, as accurately as possible, the curve calculated by the IAEA on large amount of practical data.

The formula of the activity can be used considering that *l* and *d* are the «apparent length» and the «apparent distance» of a basket. The «apparent length» takes the solid angle of the collimation device into account. The same goes for the «apparent distance» *d* which encompasses the contribution of heterogeneous composition crossing the flow.

The «apparent distance» value, used in basket shape computing, will be considered as a parameter related to the storage characteristics.

The apparent length is the same for all baskets in a silo.

[4] Monte-Carlo N-Particles transport

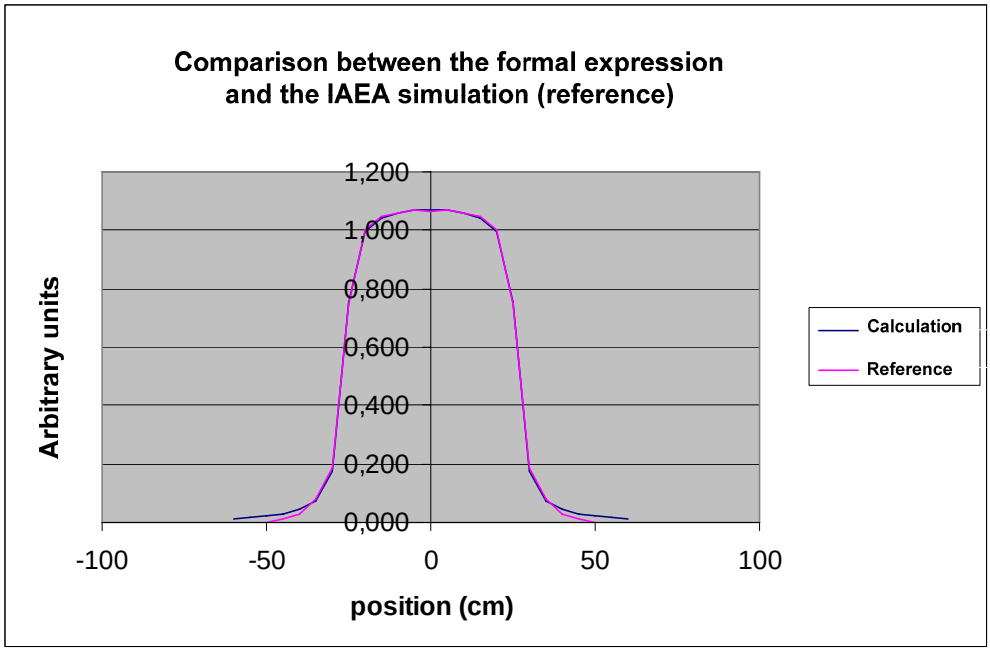


**Figure 4:** Basket model shape, matching the analytic model with MCNP simulation

### 3.3. Fine positioning of baskets centre

For noisy spectra, the method may produce some error in basket positioning. The modeling effort studied the effect of small variation of geometry parameter on correlation peak location.

A spectrum can be seen as a summation of *n=9* shapes with a noise. It can be described by:

$$S(x) = \sum_i A_i B(x, d, l, c_i) + N(x) \quad (1)$$

Where

- i is the shape number (the considered basket i= 1 to 9)
- $A_i$ is the amplitude value of the model shape
- *B* is the model shape with unitary amplitude of a basket
- *l,d* are common values (fixed) for all baskets for apparent length and distance
- x is the channel spectrum
- *S(x)* is the spectrum shape at channel x
- $C_i$ is the basket number "i" centre (fixed)
- N(x) is the noise at x

If we want to know how to detect bad positioning of the shape pattern, then the influence of first order partial derivative must be evaluated. We therefore write the 1st order development into series of the activity around $c_i$, *d and l*

.

$$S(x\ ) = \sum_i A_i [B(x, d, l, c_i) + \frac{\partial B}{\partial c_i}(x, d, l, c_i)dc_i + \frac{\partial B}{\partial d}(x, d, l, c_i)dd + \frac{\partial B}{\partial l}(x, d, l, c_i)dl + O(2)] + N(x)$$

**(2)**

This set of linear equations is simpler to solve than the non linear model shape one. This assumption is valid because we know geometric information about silo or MACSTOR which give good estimate of $c_i$, *d and l.*

To better understand let us consider a Gaussian shape as a basket model. $S_i$ can be defined by:

$$S_i(x) = e^{\frac{-(x-c_i)^2}{2\sigma_i^2}}$$

Consider two other Gaussians shifts on right and left side but still overlapping (partial derivative with respect to $c_i$) we get the following difference which is a resulting pattern. Moreover if we have a variation in $\sigma_i$, then we also have a specific figure.

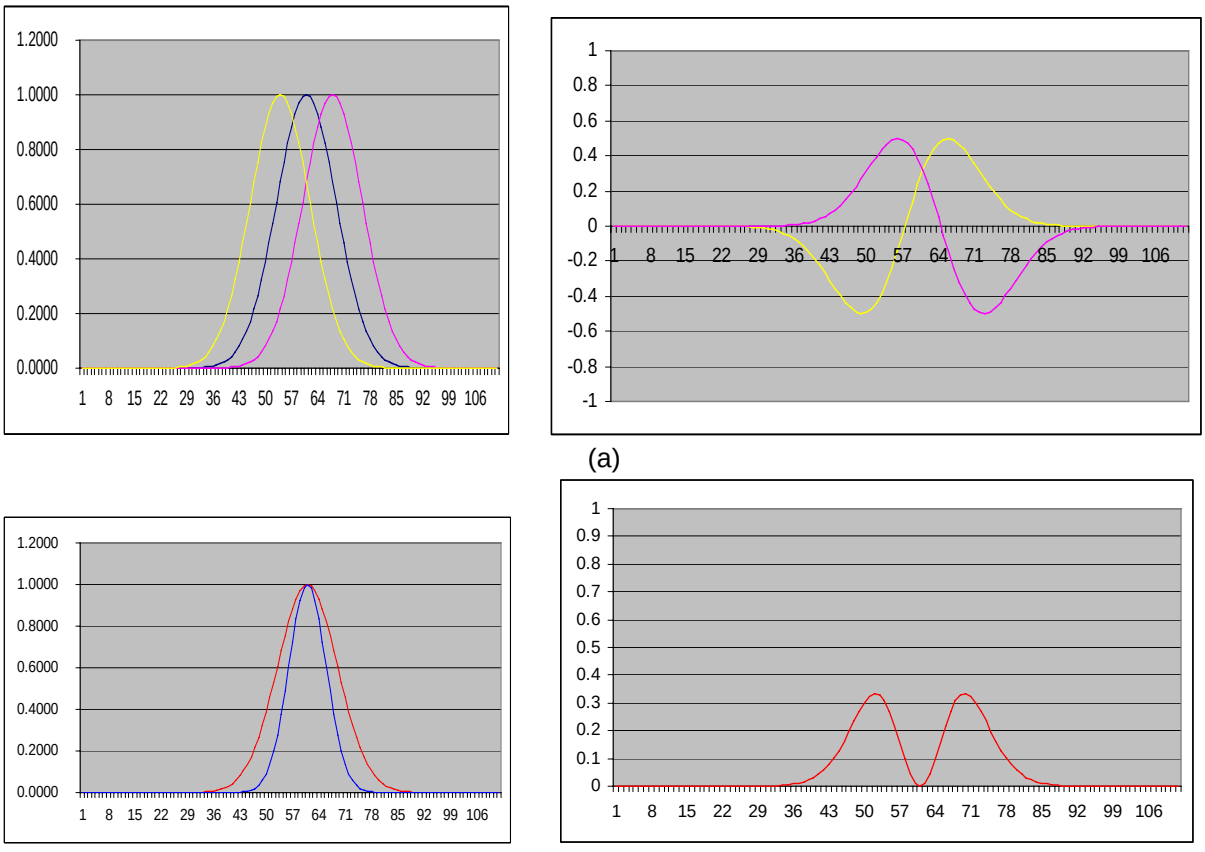


**Figure 5:** partial Gaussian derivatives according to $c_i$ in (a), according to $\sigma_i$ in (b) and resulting difference between the centered Gaussian in blue and the others.

Fine positioning of basket centre and fine tuning of basket shape are based upon the development into series at first order.

The advantage of this linearization is that we can add contribution of each basket at a given position.

The idea is then to estimate how much the parameters ($A_i$, $d$, $l$, $c_i$) must change on each basket to fit the measured spectrum.

## 7. Theoretical solution for CANDU silo and MACSTOR

Based on the model here above and since we have to discriminate against inexplicable differences, the basket localization appears to be the first step of the fitting procedure. Indeed, once localization is accurately defined we can allocate spectrum interest areas and spectrum channel to the given basket.

Therefore the spectrum processing consists in:

- Spectrum cleaning with respect to concrete rebar
- Locating the baskets
- Tuning the basket parameters
- Activity calculation
- Comparison

## 4.2 Filtering spectrum

As the measured spectrum is distorted by the concrete rebar attenuation, the location of basket can be disturbed. To suppress the contribution of the concrete rebar, the measured spectrum is first smoothed.

During the study two algorithms were tested to rub out the effect of concrete rebar.

The first one is a HPF filtering aiming at removing background noise, locating the rebar effects, removing the effect and smoothing residual irregularities.

The second one is simpler. It is a moving average method which gave good results as shown on figure 6 here below.

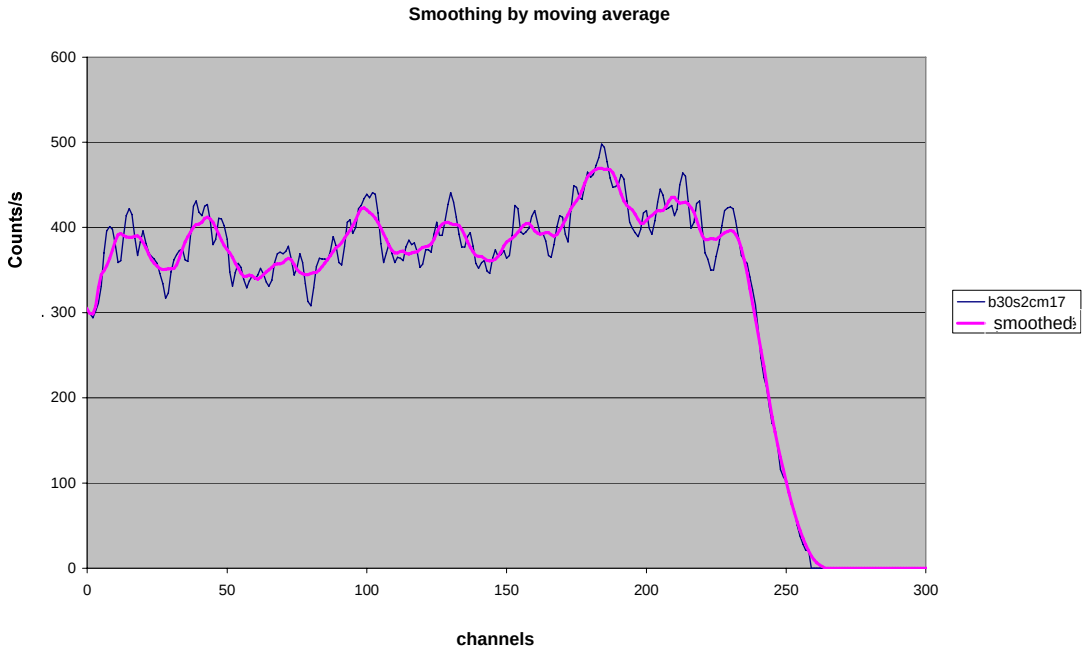


**Figure 6:** Smoothing signal with moving average

## 4.3 Basket location

The baskets positions are determined by computing a correlation between the basket shape and a window moving on the smoothed spectrum. Correlation result evolves between -1 and +1. This result doesn't depend on analyzed signal amplitude. This detection gives very thin peaks on the baskets centers (fig 7).

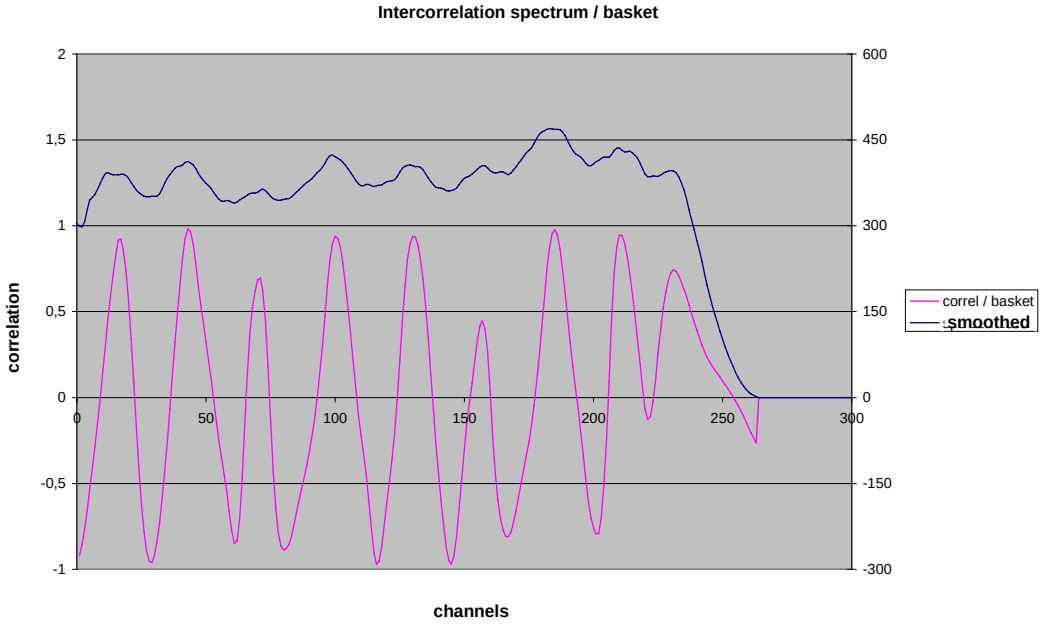


**Figure 7:** first location of basket centers with correlation

If a basket with a very low activity is located between baskets with a high activity, this basket may not be detected. If it is not detected, the found interval between the baskets is almost twice other intervals.

Before continuing the signal processing, missing baskets must be added.

Intervals between baskets are not regular. So if the probe speed motion is constant or if the z position of channels is known, the found positions of baskets can be adjusted to have a regular interval between the baskets.

However, due to the verification tube bend, the last basket, can not been taken into account. The first basket with an asymmetric spectrum due to its position must also not been taken into account in this step.

### 4.5 Parameters fine tuning

During this step, a spectrum built from the addition of activities of each detected basket is created.

Initial raw spectrum is fit as better as possible. The calculated activity is not the basket one but those attenuated by the concrete rebar. The basket shape used in this step is computed with an apparent length based on the interval between the baskets.

Following the development at first order in equation (2), the contribution of all baskets at channel i depends on three parameters:

1) $A_i$

2) $\frac{\partial B}{\partial c_i}$

3) $\frac{\partial B}{\partial l}$

The local variations of *d* and *l* have similar effects. That is why we kept only variation of apparent length *l* since it integrates the rebar contribution.

Therefore for nine baskets in a silo or a MACSTOR we have twenty seven (9x3) variables. If the measured spectrum has *n* channel we can write the linear form:

$$\begin{array}{ccccc} S & = & A & \times & X \\ \dim n & = \dim n \times 27 & & & \dim 27 \end{array}$$

with

- *S* raw measured spectrum value for the *n* channels (the observations)
- *X* the vector of 27 unknown variables.
- *A* a matrix

The activities are found by solving this system with the method of least square using the equation:

$$X = (A^T A)^{-1} A^T S$$

After parameters' fine tuning, activities $A_i$ are very accurate

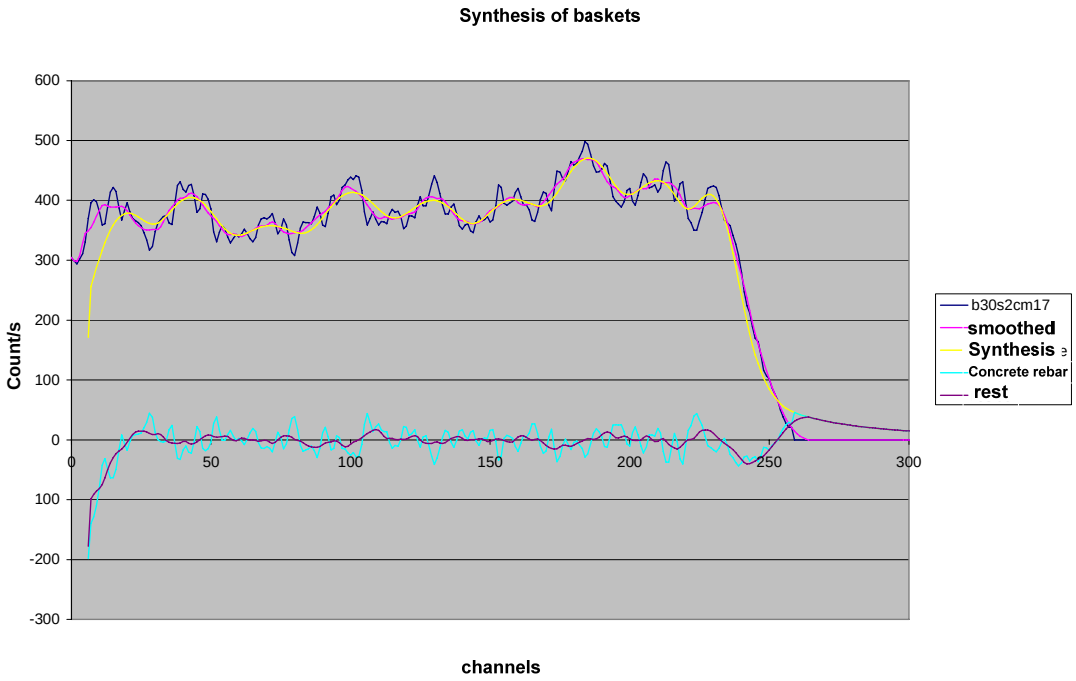


**Figure 8:** Resulting activities (Synthesis)

## 5. Modeling for DSC storage

For this dry storage no information are available to use a model based approach since there are no baskets. There are no means to improve classical spectra comparisons method in such a case. This comparison is performed on each selected areas of interest in the spectrum.

# 6. DSEF Software

## 6.1. Algorithm

For the MACSTOR or silo the algorithm provides:

- the number of baskets (with an uncertainty on the presence of the last basket)
- the precise activity of each basket. This activity is precise because it uses as best as possible the measurements in the zone situated around the center of basket.
- the uncertainty on the activity (study to be completed):
    - o theoretical uncertainty = amplitude of background noise / number of channels for 1 basket
    - o estimate of uncertainty = mean square error on the basket channels

Different measurements of the same silo with different collimators will give important deviations between the measured spectra but we will be able to compare the activities with a good confidence level.

Comparison with a baseline comes down to a comparison of scalars ($A_i$). The spectrum normalization must be done after basket area identification to avoid intermediate calculation with very small values.

If no baskets are found for MACSTOR or silo, then, a spectrum comparison channel per channel is used as for DSC.

## 6.2 Interfaces

UML with use cases method were used to design DSEF application. It brought a very powerful mean to design the application with a description level that fit the end-user requirement without entering into IT consideration.

The first purpose of the DSEF application is to store measured data over extended period of time. All stored data must be easily retrieved from the database. The measured data are stored in flat-files which can be ASCII or XML files, IEC1455File compliant such as GENIE2000 files, proprietary WinScanFile or WinMCSFile. The database therefore must store the description of each data and record the corresponding file name.

On site Inspections requires a capability to run the application on a local database (i.e. running on a laptop). Obviously this means that synchronization procedure can automatically download on-site measurement results into the DSEF system hosted on the Agency network.

The IAEA standard being based on Windows, the selected database is SQLServer 2005.

## 6.3. Using DSEF application

After having acquired new spectra, Inspector can see the result in DSEF. Then a baseline selection is possible and a comparison can be executed against four different baselines if necessary.

On the right hand side of the screen indications are given to help Inspector to understand the situation.

This is shown on the screenshots here below:

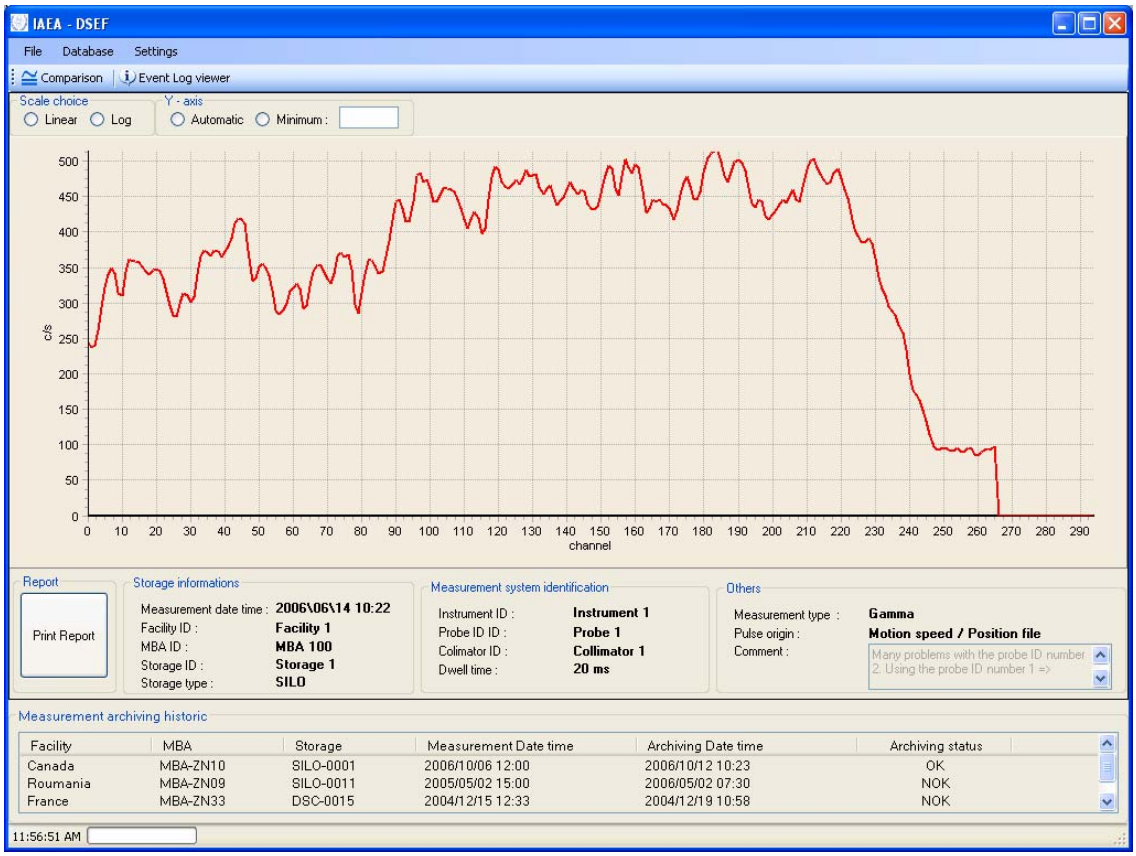


**Figure 10 a)** : Display of a spectrum store in one of the interpreted format.

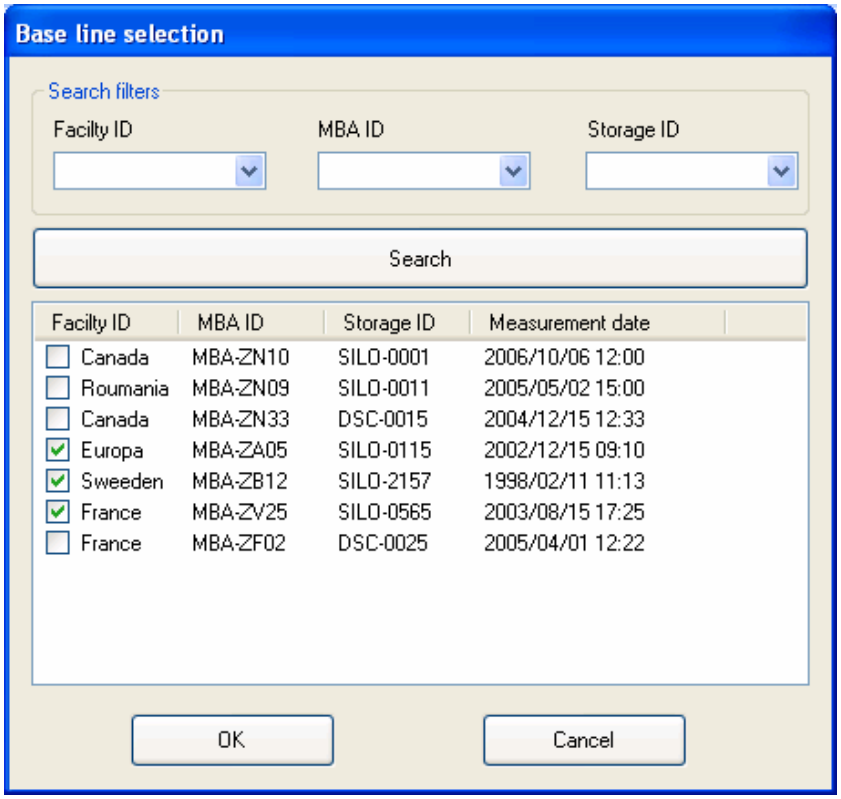


**Figure 10 b)**: Selecting a baseline

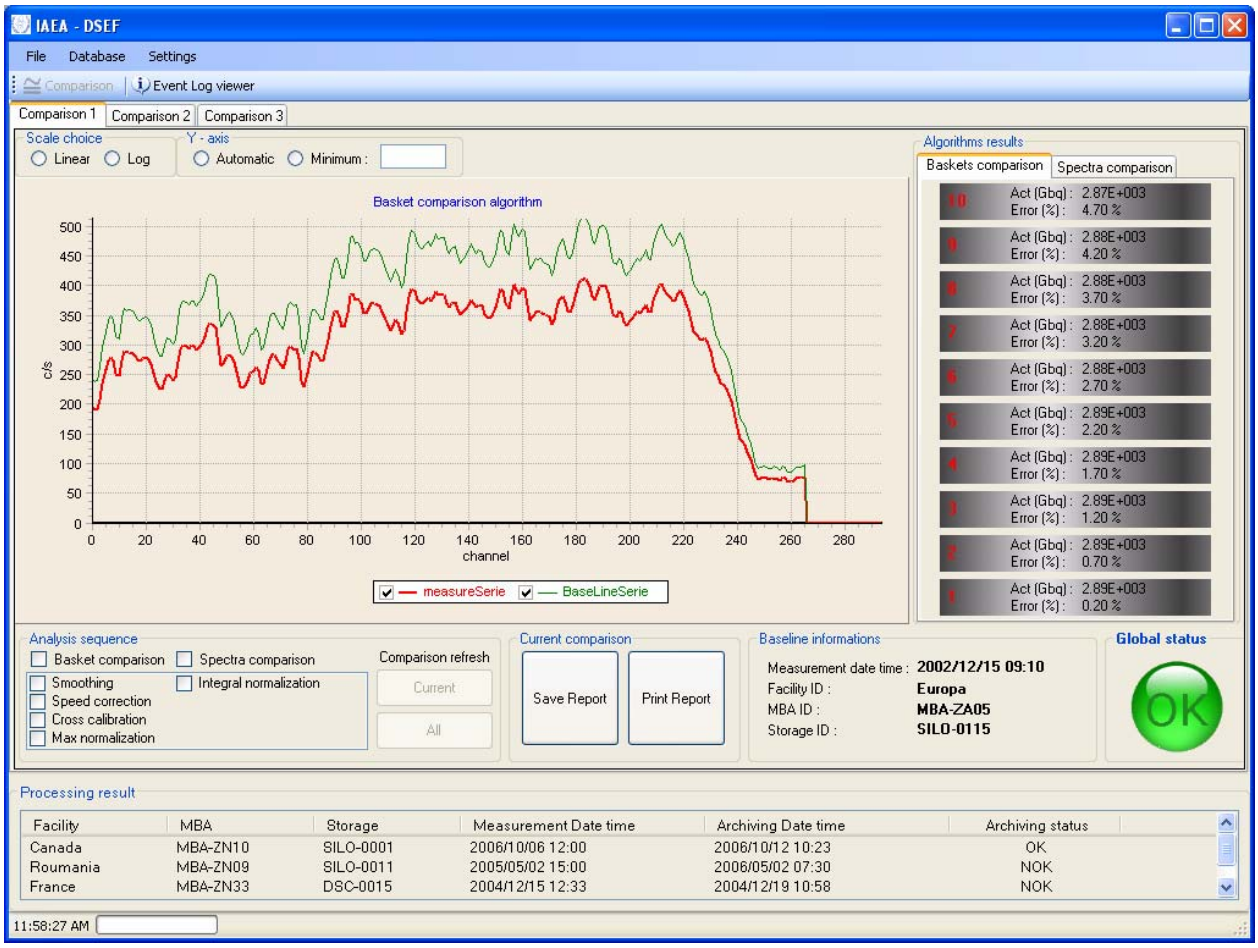


**Figure 10 c)** Successful comparisons with a baseline and with 10 baskets.

## 6.2. Testing Correlation threshold

The algorithms sensibility has been tested regarding correlation threshold, apparent length, apparent distance and basket shape length. The screenshot here below shows a case we miss a basket with a correlation threshold value of 0,9.

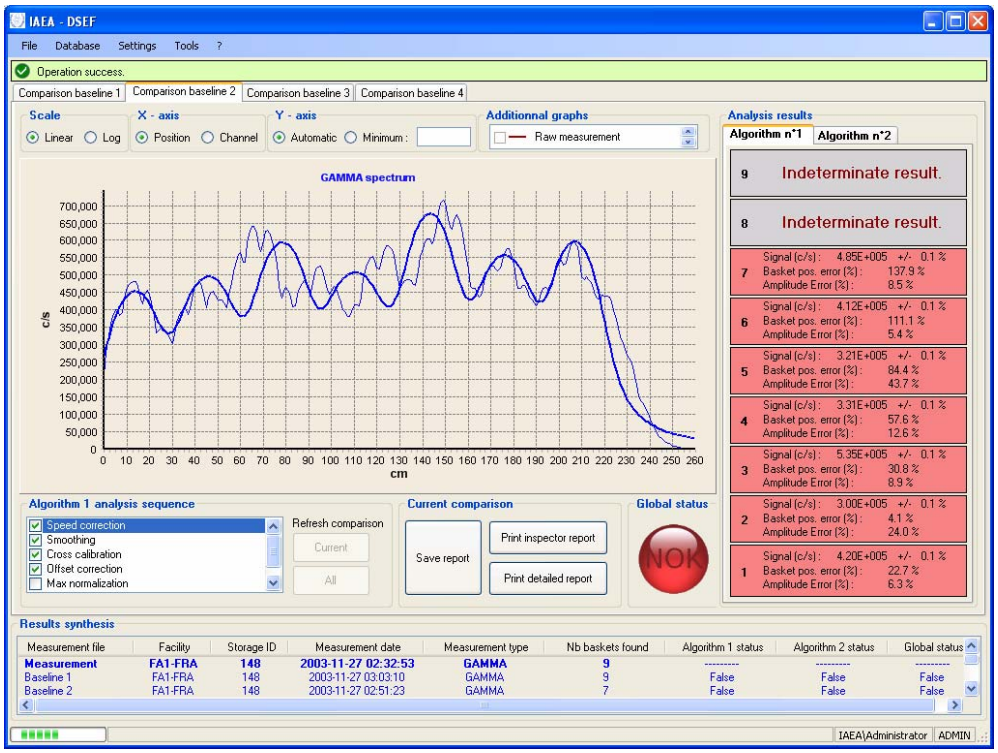


**Figure 11:** wrong value for correlation threshold

The application has a set of threshold parameters to define a good configuration (fig12)

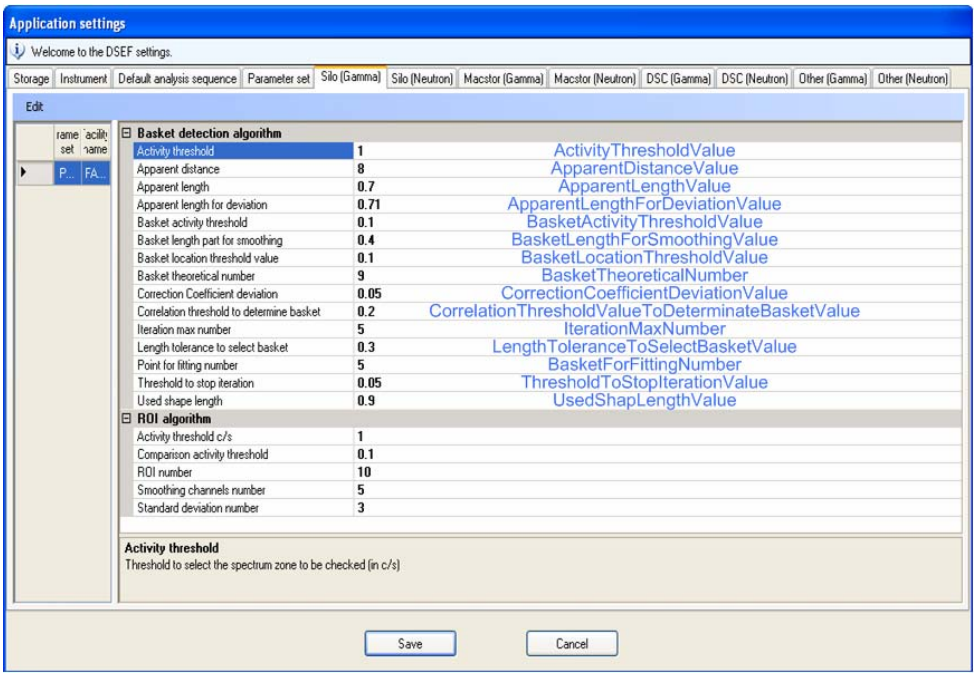


**Figure 12**: Threshold lists

## 7. Conclusion

This paper shows how we can take advantage of prior knowledge to improve algorithm. During the study we setup the problem definition in another way. The decision of looking for basket positioning before spectra matching was of great help.

Moreover this approach produces consistency control capabilities since can control the validity of results and we can raise an alarm in case of anomaly:

- on the spaces between concrete rebar (if we calculate their position)
- on the baskets number and on the spaces between the baskets
- on the consistency in comparison with the probe speed motion (measurement in t)
- on the consistency in comparison with the basket width (measurement in z).

## 8. Acknowledgement

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# Geological Repositories: Present and future verification challenges

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***Abstract:***

*The paper focuses on the verification challenges for the International Atomic Energy Agency (AIEA) in implementing integrated safeguards for the disposal of spent fuel in geological repositories. A previously developed approach under traditional safeguards is outlined and the status of current work on integrated safeguards by the Agency together with the **A**pplication of **S**afeguards **To** Geological **R**epositories (ASTOR) Group of Experts is presented.*

*The challenges related to the implementation of integrated safeguards in geological repositories are highlighted. The technical challenge, the specificity of the disposal of spent fuel in an underground matrix, the simultaneous fuel disposal and excavation work and the lifetime of the facility complicate the verification of declared nuclear material and activities. The conceptual framework developed by the Agency for safeguarding geological repositories lays the groundwork for how to take these specific factors into account. A further verification challenge lies in the technical choice to be made for verification activities and monitoring the site in order to provide assurance of the absence of undeclared activities.*

**Keywords:** Integrated Safeguards, Geological Repositories

## 1. Introduction

The IAEA has been working with interested Member States, through their support programs to IAEA safeguards, on the development of safeguards for geological repositories for many years. The IAEA's Programme for the Development of Safeguards Approaches for the Final Disposal of Spent Fuel in Geological Repositories (SAGOR) was launched in 1994 and the SAGOR Experts held a series of seven meetings, the last being in June 2004. A new phase in this development was initiated in 2006 with the formation of the ASTOR Group of Experts. The States and organizations currently participating in ASTOR are Belgium, Canada, Czech Republic, Finland, France, Germany, Hungary, Netherlands, Sweden, USA and the European Community (EC). The aim of the group is to facilitate the sharing of information between Member States and the Agency and to perform an important advisory function to the IAEA in respect to the development of safeguards approaches and techniques applicable to repositories.

The work of the ASTOR group is focused on practical aspects of the generic integrated safeguards approach for geological repositories sites and the safeguards techniques applicable to specific geological repository sites. Inter-alia it will provide the following support to the Agency:

- Review elements of integrated safeguards approaches for specific geological repositories;
- Facilitate sharing of safeguards-relevant information on geological repositories between Member States;
- Help promote understanding of safeguards requirements and implications in other technical forums and with operators of geological repositories;

- Supply experts and expertise for safeguards missions and implementation support related to repositories and
- Study and make recommendations on any issues that may arise regarding geological repository safeguards, including those mentioned above.

This paper discusses general considerations on applying safeguards to a geological repository and general principles for integrated safeguards at geological repositories, which are derived from the experience gained by the Agency in applying traditional safeguards in initiating the implementation of integrated safeguards, and from the results of the work of the SAGOR Experts.

## 2. General considerations on applying safeguards to a geological repository

Nuclear material placed in a geological repository is subject to safeguards in accordance with the applicable safeguards agreement. spent fuel does not qualify as being practicably irrecoverable at any point prior to, or following, placement in a geological formation. Therefore, the Agency does not terminate safeguards on spent fuel in a geological repository. Safeguards for such material are maintained after the repository has been back–filled and sealed, and for as long as the safeguards agreement remains in force.

The life of a geological repository is divided into three phases: pre–operational (or design and construction) phase; operational phase, and post–closure phase. According to current planning for geological repositories, the first one will reach the post-closure phase after 2060. Therefore, long-term safeguards considerations at this time can only be on a conceptual level.

On the geological repository site, there are above ground buildings and equipment and below ground areas. Above ground are provisions for storage of received containers, unloading of container contents, possibly conditioning of spent fuel including consolidation, loading of disposal canisters, and temporary storage of canisters. The below ground areas, which include access tunnel(s), ventilation shaft(s), and excavated horizontal tunnels (drifts), provide for entry and temporary storage of disposal canisters and their transfer into emplacement positions in a tunnel. For safeguards purposes, the geological formation forms the primary containment structure, restricting access to the nuclear material in a geological repository. Some of the above ground activities could possibly be located on different sites.

The objective of safeguards during all phases of a geological repository is to provide assurance of non-diversion, i.e., provide assurance that spent fuel is not diverted. The safeguards approach should take into account that once spent fuel is emplaced, it cannot be re–verified. As a consequence, sufficient redundancy, diversity and robustness should be incorporated into the safeguards system and adequate maintenance measures be applied to avoid system failure and ensure continuity of knowledge. Safeguards systems installed to function for a long period with minimum or no service, perhaps in a rugged environment, should preferably use an unattended operation mode and must meet rigorous system specifications and standards.

To meet the objective of safeguards during the entire pre-operational, operational and post closure phases of a repository, the safeguards approach should be designed to provide for verification of design information provided through the life of the geological repository, verification of the nuclear material contents of received spent fuel containers, continuity of knowledge of the spent fuel inventories above ground and below ground, and a capability to detect undeclared activities that could be associated with diversion. In order for the safeguards approach for such a repository to provide assurance that undeclared removal of nuclear material does not occur through a declared access route or through an undeclared route, it should be based on verification of receipts, nuclear material accountancy for above ground facilities, and verification of nuclear material flow between above ground and below ground areas.

## 3. General principles of integrated safeguards at geological repositories

Integrated safeguards can be implemented in a State once the Agency has concluded that there are no indications of diversion of declared nuclear material and no indications of undeclared nuclear material and activities in the State. These conclusions are reaffirmed annually through the State evaluation process, which draws on the results of activities performed by the Agency under a comprehensive safeguards agreement and an additional protocol. Integrated safeguards are implemented in accordance with an approved State-level integrated safeguards approach.

When a State has a program investigating final disposal of spent fuel in a geological repository, the State should provide the Agency with information relevant to safeguarding the spent fuel. The State and the Agency should consult in order to ensure the effective implementation of safeguards. When the program includes underground construction that may become part of the geological repository, the State should provide preliminary information on the safeguards relevant features of the underground construction, so that the Agency can initiate the preparation of a design information verification plan and make design information verification visits as appropriate. When a decision is taken to construct, or to authorize the construction of, a geological repository, the State is to provide further information in accordance with the safeguards agreement on the safeguards relevant features of the facility design early in the stages of project definition and preliminary design.

Once a geological repository is declared by a State to the Agency, the State-level integrated safeguards approach will be modified to incorporate the geological repository. The integrated safeguards approach for the geological repository and the State-level approach are modified over time as required to take account of developments at the repository, in safeguards concepts and technology, and in the nuclear activities in the State.

In applying nuclear material accountancy verification as a fundamental safeguards measure, the material balance for the nuclear material at geological repository will be reported by the State annually, and evaluated by the Agency so as to support the drawing of an annual safeguards conclusion of non-diversion. Full coverage of nuclear material flow at a geological repository will be achieved through verification of inventory changes, performed at unannounced inspections, short notice random inspections or announced interim inspections, as applicable.

Complementary access under an additional protocol to any place on the site of the geological repository, to contribute to a conclusion of no indications of undeclared nuclear material and activities, may be performed in conjunction with any design information verification visit or ad hoc or routine inspection on that site with at least 2-hour notice, or independently of a design information verification visit or an inspection with 24-hour notice.

Increased co-operation with the Regional or State’s system of accounting for and control of nuclear material (hereafter referred to as SSAC) should be considered, taking into account State-specific conditions and the technical effectiveness and capabilities of the SSAC.

In developing integrated safeguards measures to spent fuel at geological repositories, the following verification issues will be considered: the timeliness verification goal is one year; C/S measures should be applied, where applicable, at geological repositories to provide continuity of knowledge thereby increasing the efficiency of nuclear material verification. C/S measures may be used with or without remote data transmission, as decided on the basis of cost-effectiveness.

## 4. Work performed by the ASTOR Group of Experts

The ASTOR project was initiated in 2006 as follow-on to the work carried out under the SAGOR project. The ASTOR Group of Experts met for the first time in April 2006 for a kick-off meeting and a second time in October 2006. For each ASTOR technical meeting, different types of issues are addressed:

- Discussions on integrated safeguards approaches related to different elements which constitute a geological repository (e.g. encapsulation plant, transfer to the underground facility);
- Discussions on possible technical options which could enhance the conclusions of non-diversion of declared nuclear material and absence of undeclared nuclear materials and activities in a geological repository; and
- Opportunities for sharing information on the latest developments in the participating States related to geological repositories.

Between two technical meetings, discussions are further developed by the participants (through electronic communication using the Livelink tool).

Recent topics addressed to the ASTOR group are the following:

- the issues of information provided by the State to the Agency in a the preliminary phase,
- geological repository site and geological repository definition as stated in the model protocols
- the status and potential use of Ground Penetrating Radar technology for safeguards at a geological repository and;
- geological repository monitoring.

At the next meeting in June this year, it is expected to make progress on these issues, to initiate discussions of the model integrated safeguards approach for a geological repository starting with the implementation of safeguards in encapsulation plants and to discuss the application of active geophysical methods for IAEA safeguards of geological repositories for spent fuel.

## 5. Conclusions

The conceptual framework developed by the Agency for safeguarding geological repositories lays the groundwork for how to take specific factors related to these facilities into account. A further verification challenge lies in the technical choice to be made for verification activities and monitoring the site in order to provide assurance of the absence of undeclared activities. Assurance of non-diversion of spent fuel for a facility that would be underground is another challenge; this will be addressed by the use of novel technologies as re-verification of the material, once is placed in the repository, is not possible.

Finally, the continuity of knowledge for such a long-term project, as the one being discussed, constitutes a remarkable challenge for both the Agency and the Member States involved.

# Session 25

# Export control

# WATCH LISTS: Methods to reinforce export control on potentially proliferating uncontrolled items and materials

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**Abstract:**

The idea of this concept of watch list is to help proliferation and export control experts and officers to detect illicit activities. A watch list is established for the suspected country and for its preferential ways to procure nuclear materials, items and equipment. The method of establishing such a watch list is given in relation with existing export control lists and non listed items.

**Keywords:** watch lists; export control; proliferation.

## 1. INTRODUCTION

The recent attempts to get round control export rules have been revealed by affairs linked for example with the proliferating role of the Abdul Qader Khan (AQK) network in Libya, Iran and North-Korea. Such revelations make necessary to imagine new methods to counter proliferation and to fill the loopholes in the legislation.

One possible concept, the watch list concept, is to develop more elaborate and wider lists than the Trigger and Dual-Use lists [1] that are devoted to control nuclear exports. Although the present paper deals only with nuclear proliferation, the concept can be used for fighting missile, chemical and biological proliferation.

The main objectives of watch lists are to give strengthened indicators to detect proliferating countries and to screen front companies. Such a list has to be established preferentially for a given country, because reducing the number of processes that can be involved in a nuclear military program, allows a deeper control of the components and equipments of concern.

The analysis of the process that can be used by a given country has to be performed. Preliminary information on geopolitics, scientific, technological and industrial levels, and civilian existing nuclear equipments has to be reviewed by the experts. In a second term, they have to determine what path (uranium and/or plutonium ones) and in each path what kind of processes are involved, especially among the conversion processes and the uranium enrichment processes (centrifugation, laser isotope separation, calutrons…), the kind of reactors (heavy water, gas graphite...).

This control is made necessary due to the improved ways of proliferation in particular those linked to the AQK network, and the role of front companies is also to be analysed. In particular:

- Installations for the production of centrifuges components have been sold under false denomination (in the case of Libya, see for example S. Lucas and P. Louvet [2]),

- Installations, like production plants of components, have been imported like in Malaysia [2]. As a consequence, the watch lists should cover the loopholes concerning the equipment of the production plants and the elements necessary to that end.

Finally, the role played by the intangible transfers, such as numerical codes, software, technological know-how, technical assistance…), as well as second-hand items, has to be taken into consideration.

## 2. Expertise

The expertise is quite straightforward to achieve for the nuclear countries as the national nuclear entities are skilled to do it. In France, the *Commissariat à l'Energie Atomique* acts as the national expert. Primarily, the expertise has to determine which processes are aimed, as the proliferating countries usually encounters difficulties to study all the possible processes due to limitations by the huge cost and by the lack of skilled staff. Thus, such countries are obliged to focus on a limited number of processes, like centrifugation or laser for uranium enrichment, or heavy water reactors or graphite reactors for the production of plutonium. Then, a detailed functional analysis and decomposition of each process help to list the components that are expected and needed to proliferate.

## 3. Methodology to establish lists

Five main ideas are involved in establishing these watch lists:

- Filling loopholes in the legislation,
- Reinforcing the control on some "weakly" controlled technologies,
- Monitoring the use of "down-graded items",
- Taking into account the improvement of evolving technologies or new technologies,
- Including items that can be used for nuclear proliferation and that are controlled by other means.

### 3.1. Filling loopholes in the legislation for materials, subassemblies, manufacturing and inspection equipment:

The loopholes are induced in different ways. First, some items, in few numbers, have been simply omitted as they have not been judged critical or too difficult to control at the elaboration of the lists. Secondly, the loopholes are quite often related to materials, subassemblies, manufacturing and inspection equipment that are necessary to produce items or assemblies that appear in the Trigger list and Dual-Use list. Numerous examples can be found such as samarium and neodymium powders, the associated magnetization equipment, the magnetic measurement and control instrumentation to produce permanent magnets for the bearings of the centrifuge. Others cases are encountered in the key electronic components involved in the controlled power supplies or converters.
A lot of quite common chemicals that are not controlled are often needed for operating conversion, enrichment or reprocessing. They have to be added to the watch lists to draw attention for the export control officer. Some other more or less classical manufacturing equipment is also of concern.

### 3.2. Reinforcement of the control

The reinforcement of control can be achieved by increasing qualitatively and/or quantitatively the level of control in the items that are already covered by the Trigger and Dual-Use Lists.

Qualitatively, restrictions notes that are applied to the paragraphs have to be checked and modified or suppressed if necessary. A classical example is the control exemptions for medical applications.

Numerous examples have been observed for a lot of items, especially laser or laser components, materials like maraging steel or aluminium alloys, fibres…The solution is to lower the threshold values widely under the values that are needed, to be more severe and to review carefully the Trigger and Dual-Use lists.

Another possibility is to widen the field of applications to different geometries or material. For example, in order to prevent the illegal manufacturing of centrifuge rotors, maraging steels and aluminium alloys are controlled in the form of tubes in §2C11 and §2C1:
*§2C1 – « Aluminium alloys (…) with an ultimate tensile strength greater than 460 MPa (…) and in the form of cylinders or tubes (…) with an outside diameter more than 75 mm. »*
As some other minor parts can be made of these alloys in any form (rod, plate…), it is necessary to enlarge the control these materials as well. Furthermore, some proliferators can made simple metallurgic transformation using the alloys that they are not able to product to get the correct product.

Quantitatively, the main problem is raised by the numerical values that are set in the lists. The technique used for by proliferators is to buy items just under the threshold values, or to buy two or three production units instead of one with reduces performances to remain under the threshold.

**Another possibility is to reinforce the controlled of insufficiently controlled technologies.**
In this category, heavy water production technologies are globally insufficiently protected, as a number of production processes are available: G-S, ammoniac exchange, distillation, cryogenic distillation, electrolysis… Only, a few parts of these possibilities are under export control.

Another way of getting round the legislation is to ask for isotope separation devices (or parts) of other isotopes than uranium or plutonium, which are not *"especially designed and prepared for"* (EDP) and can replace or allow reverse engineering, as well as helping to develop and to optimise the processes without using radioactive nuclear materials. The main example in this category is the laser vapour isotope separation process that has been developed by South-Korea for rare-earths isotope separation (ytterbium, gadolinium) and that have been applied to make uranium isotope separation experiments [3].

### 3.3. Downgraded items

A so-called down graded item is defined here as an item that will not be classically use in an EDP item, because of its unreliability or its unappropriateness for nuclear industry, but that can be operational for a reduced life-time but sufficient to proliferate. A classical example was constituted by items like valves (but not only as vacuum quality pumps, seals, piping are also used in centrifuge plants for auxiliary circuits), that are made of stainless steel[1] instead of being made or coated with materials resistant to $UF_6$ corrosion like monel, nickel or fluoropolymers. This is possible for auxiliary circuits which involved the presence of traces of $UF_6$ in regular operations and even for principal circuits in processes that are operating with $UF_6$ at low pressure and temperature close to room one.

It can be also noticed here that second-hand items are also usable.

### 3.4. New or evolving technologies

The main example is constituted by the laser, solid state technologies and measurement instrumentation which are evolving or emerging rapidly. One of the main examples is the possible replacement of copper vapour lasers by solid laser diodes or the emergence of new types of lasers like quantum lasers.

Of course, this can be done by revising the Trigger and Dual use lists, but this huge work time consuming comes often late, although it could be started in parallel with the watch lists.

### 3.5. Inclusion of items that are controlled elsewhere

The export control lists that are published by the EU [4] include items that are controlled by other lists than the nuclear ones, i.e. chemical and biological lists originated from the Australian Group, Wassenaar list and MTCR missile list. In the EU lists, in order to avoid ambiguities, the most restraining redaction has been chosen. An obvious example to illustrate this fact is the export control of hydrogen fluoride which comes from the chemical weapons lists: this acid, essential in the cycle of production of $UF_4$ and $UF_6$ is not controlled by the nuclear Trigger and Dual-use lists. Some other

[1] The vacuum valves made of stainless steel have been added recently to the official NSG lists.

cases are much less obvious when they deal for example with lasers or lasers components on the Wassenaar list which can be applied for laser isotope separation.

Thus, although these items are already covered by the export control, it is interesting to recall them in the watch lists to the export control officers and experts as they reveal a nuclear activity on a given nuclear process and complete the scheme of export attempts and acquisitions by a proliferating country.

## 4. Conclusion

The method to draw up watch lists is quite straightforward, as the main idea is to gather inside them all that is necessary for a given country to proliferate with the processes that have been chosen. The work to elaborate such lists is important as a large number of items (hundreds in the front fuel cycle) and materials are concerned even if the number of involved processes has been voluntarily reduced. This kind of lists has to be derived specifically for non-compliant proliferating countries and acts as indicators of proliferation.

The main interest of these watch lists is to allow high reactivity, to fill the loophole with a better flexibility than with the official ones. In particular some propositions can be taken into account for export control much faster than in the classical way. Furthermore, the watch lists can give justification to apply the catch-all clause with an increase accuracy especially for countries of proliferation concern like Iran and North-Korea.

Following a French initiative, watch lists have been used as a basis and discussed with the EU member states to derive a complementary list of items in the case of Iran, which is submitted to export control in the frame of the UN resolution n° 1737 and has been published in the Official Journal of EU on April, 20$^{th}$ 2007.

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# Necessity for New Approaches and Systems to Address Proliferation Networks*

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***Abstract:***

*In recent years, the international community has learned a great deal about the extent and nature of nuclear proliferation networks and their activities. These covert activities are often spread over several States and increasingly include non-State actors. Revealing and analysing such activities poses challenges to the IAEA that can only be solved by developing new approaches and systems.*

*This paper describes the latest IAEA efforts to develop approaches, methods and systems for handling and analysing data related to proliferation networks. The goal of these efforts is to provide early indications of possible undeclared nuclear activities. This will enhance the State evaluation process and further strengthen international safeguards.*

**Keywords:** safeguards, knowledge management, outreach initiative, proliferation networks, trans-State, non-State, sub-State

## 1. Introduction

The role of traditional safeguards is to verify State declared information using methods such as destructive analysis and non-destructive assay (DA/NDA) techniques, surveillance, and checking material accounting reports related to declared facilities and locations outside facilities (LOF), based on comprehensive safeguards agreements (CSA) between the IAEA and States.

In 1991 and 1992, when Iraq's and North Korea's clandestine nuclear programmes were revealed, it became clear that signatories to the Non-Proliferation Treaty (NPT) could, and would, divert from their agreement for several reasons such as external threats, national and international prestige or achieving technical superiority. These revelations were the first clear indication of the changes in proliferation threats that face the IAEA.

## 2. The proliferation threat changes

Safeguards verification through CSA is concerned with diversion of nuclear material done by State actors. However, the last decade has revealed new challenges to safeguards. Proliferation activities uncovered were the result of trans-State activities performed by non-State/sub-State actors such as deviant political groups, or corporate entities simply trying to make money by supporting a nuclear programme. Such activities were a major shift in proliferation activities that the IAEA had to respond to. Understanding the nature of the actors and activities involved is the first step towards developing methods and tools to detect them.

*Non-State actors* are entities that are not associated with the State authorities and act within a State without the State's official knowledge. The Tinner family members' activities are a good example of

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such entities, where their activities, although conducted on Swiss soil, did not involve knowledge by the Swiss authorities.

*Sub-State actors* are similar to non-State actors with one difference: they are associated with the State authorities. Dr. A.Q. Khan in Pakistan and his associates were prominent members of the Pakistani scientific community and were associated with the State authorities. However, they formed one of the most daring nuclear proliferation networks. They were using a State facility, the Khan Research Laboratory (KRL), to support their activities that were possibly carried out without the Pakistani Government's official knowledge.

*Trans-State activities* are proliferation activities that take place across more than one State for the benefit of a State or a group. The declared Libyan attempt to acquire nuclear technology involved activities that were carried out across several States by a combination of non-State and sub-State actors for the benefit of one State: Libya.

Modern industries and technologies pose yet another challenge. Advanced technologies, materials and know-how are available to wider user domains. This means that proliferators may have a better opportunity to satisfy the needs of a clandestine programme without detection by export controls. The challenge is to detect the nuclear capable trade incidents, regardless of the export control thresholds, and highlight them.

## 3. Response to new challenges

A major IAEA goal is to deliver soundly based safeguards conclusions thus providing credible assurances to the international community of the peaceful use of nuclear energy, and that States are fulfilling their safeguards obligations. To reach this goal and to address the risk of nuclear material diversion, the IAEA started seeking wider competences to be able to apply stronger safeguards measures to verify not only the correctness but also the completeness of State declarations.

In February 1993, the IAEA Board of Governors (BoG) endorsed a Voluntary Reporting Scheme (VRS)[1] on imports and exports of nuclear material and exports of specified equipment and non-nuclear material, as an early measure for the IAEA to improve its information sources.

In May 1997, the BoG approved a model Additional Protocol (AP) to safeguards agreements[2], to declare not only the States' nuclear materials, but also their research and development activities, planned facilities related to or supporting the national nuclear programme, and exports and imports of controlled single use equipment. The AP gave the IAEA more power to investigate activities closely related to the nuclear fuel cycle.

The VRS declarations, together with those of the AP, enabled the IAEA to have a wider view of the transfer of single use equipment and materials, and strengthened its capabilities to detect diversion activities within a State and to verify the completeness of the State's declarations. However, the improved access and export-import declarations still covered activities in declared locations only.

Covert nuclear proliferation networks impose the need for changes in safeguards strategy to detect them. These changes are reflected in the IAEA Medium Term Strategy 2006-2011 as a need to "develop and/or use new concept approaches, techniques and technology for information analysis and verification activities, especially with regard to enhanced ability to detect undeclared nuclear material and activities" and to "Obtain, through appropriate mechanisms and channels, pertinent information on international nuclear activities and trade relevant to safeguards implementation[3]".

A trans-State analytical approach, where nuclear trade relevant information received from all States and sources is cross referenced, could provide a broader overview of the proliferation indicators worldwide, and the possibility to detect proliferation attempts early enough for timely action.

## 4. New analytical approaches

In 2005, the IAEA General Conference[4] invited all States to cooperate with the IAEA in its efforts to verify and analyse information provided by Member States on nuclear supply and procurement. The IAEA's Secretariat is reaching out to States that might be willing to help the Secretariat in this regard, with a view to gaining access to information that could strengthen the IAEA's safeguards system. The provision of such information to the Secretariat is done entirely on a voluntary basis and is handled with the highest level of confidentiality.

The IAEA's procurement outreach initiative is based on the premise that entities related to covert networks are likely to leave visible traces, as they try to acquire nuclear-related goods and services on the open market. Procurement outreach is designed to acquire access, with the agreement of the States concerned, to such traces.

On this basis, the IAEA intends to agree on appropriate modalities with Member States for access to safeguards relevant trade related data that is not normally available to the IAEA. It should be underlined that such information is sought in order to support the IAEA's nuclear verification mandate, which does not encompass export control per se.

Companies in selected business sectors would be encouraged to watch for procurement enquiries received from entities seeking to acquire goods that might be included in a nuclear programme. It is a combination of several features, each innocuous in itself, that identifies a suspicious enquiry - not all features are related to the goods being sought.

Outreach information received is mainly trade related and presents quite new challenges to the IAEA information systems. The data varies in reliability, nature, sensitivity and structure and its handling requires a change in understanding of the institutional memory. This information, combined with appropriate analysis, can provide early indications of attempts to circumvent States' safeguards and other nuclear non-proliferation undertakings[5, 6].

## 5. Providing the right tools for analysis and data handling

In the past, the IAEA received highly structured State declared information. This information had well known and expected format, and was as reliable as the State producing it. Today, in addition, the IAEA is receiving new information, which is amorphous in nature and often incomplete, varying in accuracy and content. Further, the resulting information overload imposes even more challenges to the IAEA. The amount of information currently in the IAEA's possession is very large and is expected still to grow in the future.

For all types of covert nuclear trade related data, the IAEA must maintain an efficient institutional memory documenting past and current nuclear trade related activities and concerns. This is increasingly important with proliferation networks. Most of the currently known proliferating actors have a track record of proliferation in the past. Archived records can prove to be invaluable in current and present investigations and analysis.

The IAEA has developed and is using the Procurement Tracking System (PTS) for handling and archiving all available information from past and current safeguards relevant nuclear trade activities. PTS is an information management system that is designed to guide the analytical process by associating disparate facts from multiple sources and attaching quality judgments to data. Low quality data will not be discarded, rather analysed, flagged and saved. Information and associations of facts are recorded in a way that alerts analysts to trends and associations, with modern visualisation software in place. Another function of the tools is to allow new analysts to access past knowledge of activities outside of their experience.

PTS incorporates several data mining and analytical tools, the function of which is to aid analysis and recognize networking patterns by blending new information with that already stored in the institutional memory thus allowing safeguards to react more promptly on information. It is designed with quality management in mind, to produce repeatable results. The cycle of "Do-Check-Act" is well incorporated in the system to ensure the reliability and quality of the decisions taken based on it[7].

The system has a document control module to record the source and the reliability of each document. All documents are added to a folder-structured file storage module designed to optimize the time required to reach any file. In addition, this module is fully indexed and searchable in an attempt to decrease the time required for identifying a relevant document. Further, a visualization system[8] aimed at visualising unstructured text is attached to this module providing a visual view to vast amount of search results and identifying relevant documents even faster.

In addition, PTS has a structured text module based on a relational database to record information and its reliability. This database is optimized to uncover unknown links between entities, discovering possible proliferation networks. Attached to this module is a visualization tool aimed at visualising structured data from the relational database. This tool helps the analyst to understand the complex nature of proliferation networks by visualizing the relationships between entities, which is not efficient if performed only in textual format.

In an effort to minimize the time required for data entry, the IAEA is looking into establishing an automated data extraction and categorization engine. This engine will feed the relational database with data extracted from the file storage. It will also analyse the documents and identify their relevant categories. These categories will be presented to the user in a virtual folder structure. This will enable the analyst to browse the contents of the institutional memory by items of interest in a fully automated manner.

PTS and its tools are being further developed with the support of IAEA Member States Support Programmes and are described in more details elsewhere[9].

## 6. More approaches are still to be developed

Studying past and current proliferation activities shows that each case is unique in the factors affecting it. There is a need to establish objective and quantitative evaluation methods, taking into account, among others, political, tribal, security, religious, financial, social and psychological factors that might lead to proliferation.

These methods would contribute to risk assessments to enable the IAEA to better understand possible proliferation trends in States and regions. In addition, it would provide the IAEA with an independent, systematic evaluation method that would enable further improvement of the State evaluation processes.

## 7. Conclusions

The IAEA aims, among other things, to conclude that there are no undeclared nuclear materials or activities related or leading to covert nuclear developments anywhere in a State. Undeclared activities may be State-, non-State-, or sub-State- sponsored, or a result of trans-State activities. Information related to such activities and actors require additional handling approaches.

The IAEA is also aiming to verify that activities spread across more than one State (trans-State) are not related or leading to covert nuclear weapons development, and is developing additional analytical approaches and tools to support these verification efforts. A trans-State analytical approach being developed is addressing these needs.

New information sources, such as those resulting from the outreach initiative, are essential to detect and understand proliferation networks and to verify the completeness of State declarations. Information systems, such as PTS, are helping IAEA to deal with the information overload and assisting analysts in their tasks to understand the extent of known proliferation networks and to uncover currently unknown ones early enough.

Further, States should not only be evaluated individually, but also in a regional context. There is a need for developing additional methods to evaluate factors, other than nuclear materials and technologies, which might indicate that a State is considering a covert nuclear programme.

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# Open source information for export control

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***Abstract:***

*This paper presents some preliminary work in the area of export control for nuclear proliferation. The long term goal is to access and analyse data that may relate to the trade of nuclear dual-use items and technologies: this data is available through various open sources. The immediate goal is to work on a table of correspondence between codes that describe, on one side, nuclear items and the nuclear- dual-use, and, on the other side, goods as they are declared in trade databases. We present a first exploration on the use of language technology to help bridge these two, very different, worlds of terms.*

**Keywords:** export control, dual-use, combined nomenclature, language technology, open source.

## 1. Introduction

Non-proliferation agreements are inter alia reflected under export control regimes: these pose restrictions on the trade of items which can assist the manufacturing of chemical, biological or nuclear weapons or other nuclear explosive devices as well as missile technology.
In the context of nuclear non-proliferation, the **Nuclear Suppliers Group** (NSG) authored specific guidelines describing items of interest to nuclear proliferation and aiming at controlling the export of nuclear material, equipment and technology (INFCIRC/254/Part 1) [1] as well as for the transfer of nuclear-related dual-use equipment, materials, software and related technology (INFCIRC/254/Part 2) [2].
At European level, the NSG guidelines are incorporated as part of the broader **Council Regulation 1334/2000** [3] and amendments [4]. This single regulation implements four internationally agreed dual-use controls, namely the Wassenaar Arrangement, the Missile Technology Control Regime (MTCR), the Australia Group the Chemical Weapons Convention (CWC) and the Nuclear Suppliers Group. The regulation identifies a single list of items to be controlled over the four Regimes: items are enumerated by a coding scheme, here referred to as **'EU dual-use codes' (EU-DU-C),** whose structure and granularity is driven by proliferation concerns.

Databases on international trade present an interest in export control. At European level, COMEXT [5] by EUROSTAT collects data on trade between EU Member States and non-member countries. Another example is COMTRADE [6]: maintained by the Statistics Division of the United Nations, it provides a worldwide view on trade. **In these databases trade data is reported according to product classification schemes that are independent from EU-DU-C.** For instance, in COMEXT items are indexed by **'Combined Nomenclature codes' (CN-C)** [7] whose structure and granularity reflect customs tariffs and not proliferation concerns.
**As a result, to access trade data relevant to nuclear non-proliferation, it is first necessary to establish a mapping between the various sets of codes, e. g. EU-DU-C to CN-C.**

The paper reports on exploratory work carried out on the official EU **'Correlation Table' that maps EU-DU-C to CN-C, with a focus on items identified in the NSG guidelines.**
The first goal of the exercise is to evaluate and eventually improve the quality of this correspondence table by coupling nuclear domain expertise with language technology tools.
The longer term objective is to use the table as a key to access trade data for verification; in doing so, we are aware that any mapping between coding schemes inevitably introduces approximate queries, but these approximations cannot be avoided. Exporters of dual-use items and custom officers are both

| CN-C | EU-DU-C |
|---|---|
| 84011000 | 0A001a |
| 84014000 | 0A001b |
| 84261100 | 0A001c |
| 84261900 | 0A001c |
| ... | ... |

**Table 1:** First lines the EU Correlation Table**.**

confronted with the same dilemma: traders are required to attribute CN codes to their exports (EU-DU-C to CN-C); vice versa, customs officers need to identify the nature of exports from CN codes (CN-C to EU-DU-C). They both use the Correlation Table, each in the appropriate direction.

## 2. The official EU Correlation Table

At EC level, DG Taxation and Customs Union (TAXUD) maintains the official EU Correlation Table that creates a correspondence between combined nomenclature codes (CN-C) and EU dual-use codes (EU-DU-C). Updates of the table reflect amendments of the Council Regulation on dual-use [3, 4] as well as the yearly revision of the Combined Nomenclature [7].
The mapping provided by the table is a not a one-to-one mapping: the same CN-C may describe more than one EU-DU-C; likewise, several CN-C may be associated to the same EU-DU-C.
Few entries of the Correlation Table are shown in Table 1. The first line, for instance, associates:

*Nuclear reactors [Euratom] (84011000)*

and

*Nuclear reactors capable of operation so as to maintain a controlled self-sustaining fission chain reaction (0A001a)*.

To date, the Correlation Table serves the practical purpose of informing exporters and custom officers of Member States on the restrictions that apply to the trade of goods defined in the Council Regulation on dual-use [3, 4].
Technically, the Correlation Table is part of the 'Integrated tariff of the European Communities' (**TARIC** [8]). TARIC incorporates the Community legislation on trade concerning tariff suspensions, quotas, import/export prohibitions, surveillance, restrictions, etc. It identifies goods by TARIC codes, a subdivision of CN-C codes that adds 2 rightmost digits.
TARIC is used by the Commission and the Member States for the purpose of applying Community measures relating to imports and exports. A web site [9] is dedicated to the consultation of TARIC (Figure 1, left). For example, by entering the TARIC code 8401100000 (corresponding to 'Nuclear reactors') an exporter is made aware of the restrictions that apply to the trade of this category of goods (Figure 1, right): namely, that an export authorization is required because of Regulation R0394/06. The footnote CD464 also gives access to EU-DU-C corresponding to 8401100000 (i.e., 0A001a, not

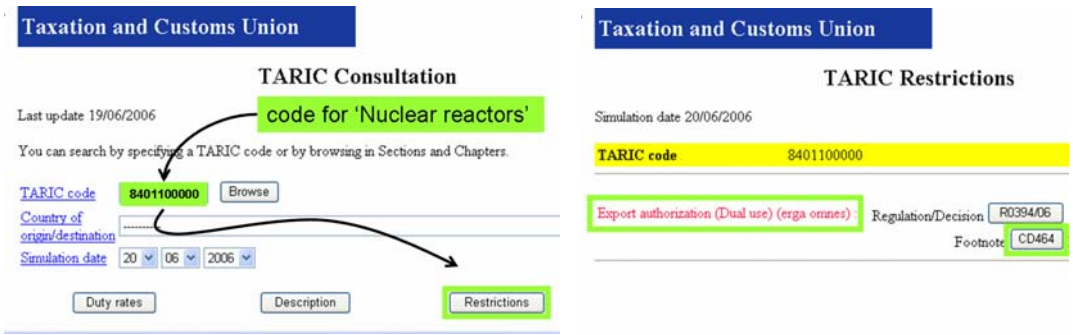


**Figure 1:** The TARIC consultation site provides information about trade restrictions.

| Nr | Dual-use Code | CN Code | Meaning of CN Code |
|---|---|---|---|
| 1 | 0A001a | | |
| | | 8401_10_00 | Nuclear reactors [Euratom] . |
| 2 | 0A001b | | |
| | | 8401_40_00 | Parts of nuclear reactors, n.e.s. [Euratom] . |
| 3 | 0A001c | | |
| | | 8426_11_00 | Overhead travelling cranes on fixed support . |
| | | 8426_19_00 | Overhead travelling cranes, transporter cranes, gantry cranes, bridge cranes and mobile lifting frames (excl. overhead travelling cranes on fixed support, mobile lifting frames on tyres, straddle carriers and portal or pedestal jib cranes) . |
| | | 8426_99_00 | Ships" derricks; cranes, incl. cable cranes (excl. overhead travelling cranes, transporter cranes, gantry cranes, portal or pedestal jib cranes, bridge cranes, mobile lifting frames and straddle carriers, tower cranes, works trucks fitted with a crane, mobile cranes and cranes designed for mounting on road vehicles) . |
| | | 8428_90_97 | Lifting, handling, loading or unloading machinery, n.e.s. . |
| 4 | 0A001d | | |
| | | 8401_40_00 | Parts of nuclear reactors, n.e.s. [Euratom] . |

**Figure 2:** Extract from the Nuclear Correlation Table with the meaning of CN codes added.

shown in Figure 1).

## 3. Focusing on the Nuclear Correlation Table

The Correlation Table covers the whole Council Regulation on dual-use which is broader than the nuclear focus offered by the NSG guidelines.
Related to this, it is to be noted a *different use of terms within the nuclear community and between those who refer to the Council Regulation on dual-use*. For the nuclear community, NSG Part1 is purely nuclear (i.e., it includes "nuclear" material, equipment designed for nuclear industry and declared as such) and it is also referred to as 'trigger list'[1]. NSG Part2 covers those equipment that can be used inter alia for nuclear explosive activity, hence the qualification of 'dual-use' equipment. On the other hand, the Council Regulation does not make this distinction and calls dual-use items appearing both in NSG Part1 and Part2. Even more, all items listed in the Regulation are called dual-use and they derive from four export control regimes that originated independently and intersect on some items.
As a consequence, if one is primarily interested in trade data related to NSG Part 1 and Part 2, three steps need to be accomplished:

1. Tag items of interest in the NSG guidelines using the native NSG-C coding system;
2. Retrieve these items within the Council Regulation on dual-use, i.e. establish the mapping NSG-C → EU-DU-C.
3. Use this mapping to identify within the EU Correlation Table all and only the lines that relate to NSG-C: these lines together make the **Nuclear Correlation Table**. This is equivalent to derive the correspondence NSG → CN-C.

Figure 2 shows part of the Nuclear Correlation Table sorted by EU-DU-C. The meaning of each CN code (derived from the 'Structure and self-explanatory texts' of the Combined Nomenclature downloadable from [7]) has been added for clarity.
Table 2 provides numbers on the size of the Nuclear Correlation Table in comparison to the complete Correlation Table.

| Correlation Table | Number of rows | Distinct EU-DU-C | Distinct CN-C |
|---|---|---|---|
| Complete | 3189 | 548 | 928 |
| Nuclear | 571 | 116 | 372 |

**Table 2:** Comparing the size of the complete and of the nuclear Correlation Table.

[1] One should note that import / export of "nuclear" materials are addressed by the Euratom Treaty and derived regulation.

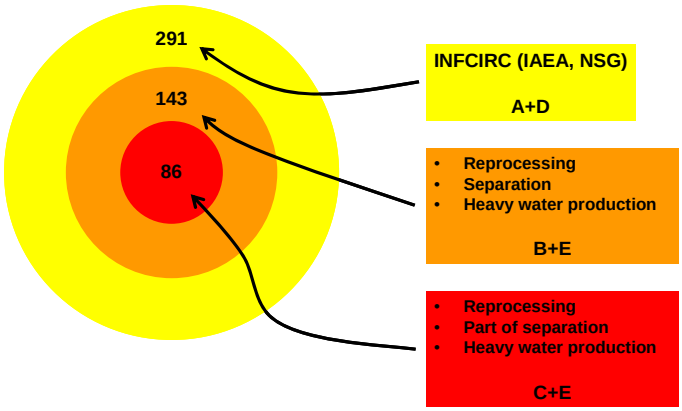


**Figure 3:** Priority levels applied to items in the NSG guidelines.

## 4. The 'size' of the nuclear export control problem

To have an idea of the 'size' of the nuclear and nuclear-related export control problem as depicted by the NSG guidelines, we counted the items listed in these documents.

- The total number of items amounts to 291.
- Of these, 133 come from NSG Part 1 [1].
- 158 come from NSG Part 2 [2].

A priori, all NSG items are relevant for the nuclear export control problem. Nevertheless, to manage the size of the problem, we can identify **different levels of attention on items** pertaining to processes of the fuel cycle that are more critical or appropriate with respect to nuclear proliferation. Indeed, regarding enrichment, not all isotope separation methods are equally interesting from the proliferation point of view.
The following **prioritization of NSG items** is proposed.

Three levels of priority A, B and C are defined on NSG Part1.

- Level **A** includes all items → **133 items**.
- Level **B** includes 'Equipment and Non-nuclear Materials' items that fall in sections 3. (Reprocessing) and 5. (Separation of isotopes of uranium) → **98 items**.
- Level **C** includes 'Equipment and Non-nuclear Materials' items that fall in sections 3. (Reprocessing) and sections 5.1., 5.2. and 5.7. → **41 items**.

Two levels of priority D and E are defined on INFCIRC/254/Part 2.

- Level **D** includes all items → **158 items**.
- Level **E** includes items that fall in sections 3. (Uranium isotope separation equipment and components) and 4. (Heavy water production plant related equipment) → **45 items**.

For illustration purposes, Figure 3 gives a view on how these prioritized items add in numbers by combining:

- A and D → **291 (yellow)**
- B and E → **193 (orange)**
- C and E → **86 (red)**

'Yellow' covers the complete NSG guidelines, Part 1 and Part 2. 'Red' are the core items one may want to control with the highest level of attention. 'Orange' is an intermediate level.

## 5. Language technology and the Nuclear Correlation Table

There are two motivations for implementing language technology in relation to the Nuclear Correlation Table.

| 1 | TERM | LH_MEASURE | 15 | centrifuge | 105,1636299 | 29 | vapour | 62,38738146 |
|---|---|---|---|---|---|---|---|---|
| 2 | resistant | 313,5056238 | 16 | equipment | 90,71433966 | 30 | suitable | 56,78168827 |
| 3 | uranium | 312,4852022 | 17 | tantalum | 87,59108827 | 31 | collector | 56,71265678 |
| 4 | corrosion | 279,0118061 | 18 | fluorocarbon | 87,59108827 | 32 | frequency | 56,24154641 |
| 5 | separation | 275,6854046 | 19 | design | 86,03753292 | 33 | exchanger | 54,32328568 |
| 6 | protect | 135,3947083 | 20 | laser | 85,57661264 | 34 | concentrated | 54,11391155 |
| 7 | specially | 132,9561461 | 21 | rotor | 84,92623772 | 35 | magnet | 51,3371621 |
| 8 | diameter | 128,5308539 | 22 | component | 73,72527024 | 36 | crucible | 50,916371 |
| 9 | graphite | 116,2967447 | 23 | nozzle | 70,74382037 | 37 | plant | 50,88764534 |
| 10 | ion | 111,5755067 | 24 | cryogenic | 70,01370589 | 38 | compressor | 50,67530203 |
| 11 | gas | 110,7547625 | 25 | hydrochloric | 69,38046152 | 39 | gaseous | 49,56832132 |
| 12 | tube | 109,9768268 | 26 | therefor | 65,41115518 | 40 | temperature | 49,55252851 |
| 13 | isotope | 107,4537651 | 27 | ion-exchange | 63,96860447 | 41 | cylindrical | 46,05670189 |
| 14 | capable | 106,6827657 | 28 | consist | 63,45344924 | 42 | heat | 44,28932461 |

**Figure 4:** Ranking of single words by their significance for the paragraph 0B001 of the Council Regulation.

First, before using the Table to access trade databases, we want to be in a position to assess the quality of the mapping between EU-DU-C and CN-C. The issue is to establish to what extent this mapping is accurate and whether certain parts of it need and can be improved. In general, answering these questions requires deep knowledge of the Combined Nomenclature. Nevertheless, since a textual description is attached to both EU-DU-C and CN-C, we use language technology to match these descriptions in an automatic way, i.e. without the help of experts of the Combined Nomenclature.

Second, considering that both the Council Regulation and the CN exist in all 23 official languages of the EU, using language technology opens the possibility to perform this evaluation in the languages of preference of the users, possibly mixing languages in a team.

The idea underlying the language technology approach followed here is to suggest associations between EU-DU-C and CN-C by focusing on items referred by the same **words of significance** (see paragraph 3.1 in [10]) **both in the EU-DU-C and CN-C textual descriptions**. Then, processing the results in an appropriate way should help to confirm, refine and perhaps make more precise the correspondence between the two sets EU-DU-C and CN-C.

Potentially relevant terminology (significant words) from the field of Nuclear Non Proliferation (NP) was identified in two different ways, one for single words and another one for multi-word terms. For single words, a statistical method was used to identify which words are statistically significantly more frequent in NP documents compared to general documents. For this purpose a frequency list of words in a collection of NP documents was produced and compared to a generic word frequency list, using the standard 100 Million words British National Corpus BNC [11]. Both frequency lists were compared using the log-likelihood test. This test produces a ranked list of words that are surprisingly frequent. Figure 4 shows an example, the result obtained by ranking the words that describe items in section 0B001 of the Regulation.

To identify multi-word terms that are typical for NP-related texts, we first applied linguistic patterns to the NP texts to select noun phrases such as ‘Word1 Word 2’, ‘Word4 Word4 Word5’ or ‘Word6 Word 7 Word8 Word9’ and we then used statistical methods to select those that seemed most typical for texts from the NP domain. For this purpose, we used the “Tree Tagger” [12] software to recognise the part-of-speech of words (noun, verb, adjective, preposition, etc.) and then filtered out noun-noun or adjective-noun sequences, etc. As the resulting list contains very common noun phrases such as “current output” as well as specialist terminology such as “isotope separation”, we applied various statistical measures (Mutual Information, log-likelihood test, etc.) to determine which combinations are statistically outstanding.
The overall results were thus NP-related lists of single words or compound expressions.

Having determined our words of significance, we illustrate how to use them to retrieve CN/TARIC items in correspondence to dual-use items of interest.
In this example we focus on item 0B001b11 whose textual description in the Council Regulation is:

*Centrifuge housing/recipients to contain the rotor tube assembly of a gas centrifuge, consisting of a rigid cylinder of wall thickness up to 30 mm with precision machined ends and made of or protected by ´´materials resistant to corrosion by UF6´´;*

where the words of significance appear underlined. They are displayed below according to the order of apparition in the text. In parenthesis, we have indicated their ranking by relative frequency, as explained above and as shown in Figure 4.

- centrifuge (3)
- gas centrifuge
- rotor tube assembly
- rotor (4)
- wall thickness
- cylinder (5)
- resistant (1)
- corrosion (2)

**Goods: centrifuges**

| | Taric code | Description |
|---|---|---|
| | **8421** | **Centrifuges, including centrifugal dryers; filtering or purifying machinery and apparatus, for liquids or gases** |
| 1 | 8421000000 | Centrifuges, including centrifugal dryers; filtering or purifying machinery and apparatus, for liquids or gases |
| 2 | 8421110000 | Centrifuges, including centrifugal dryers |
| 3 | 8421192000 | Centrifuges of a kind used in laboratories |
| 4 | 8421910000 | Of centrifuges, including centrifugal dryers |

**Figure 5:** TARIC search hits on ‘centrifuges’.

**Goods: centrifugal**

| | Taric code | Description |
|---|---|---|
| | **8413** | **Pumps for liquids, whether or not fitted with a measuring device; liquid elevators** |
| 1 | 8413700000 | Other centrifugal pumps |
| 2 | 8413708100 | Other centrifugal pumps |
| | **8414** | **Air or vacuum pumps, air or other gas compressors and fans; ventilating or recycling hoods incorporating a fan, whether or not fitted with filters** |
| 3 | 8414594000 | Centrifugal fans |
| | **8421** | **Centrifuges, including centrifugal dryers; filtering or purifying machinery and apparatus, for liquids or gases** |
| 4 | 8421000000 | Centrifuges, including centrifugal dryers; filtering or purifying machinery and apparatus, for liquids or gases |
| 5 | 8421110000 | Centrifuges, including centrifugal dryers |
| 6 | 8421910000 | Of centrifuges, including centrifugal dryers |
| | **8450** | **Household or laundry-type washing machines, including machines which both wash and dry** |
| 7 | 8450120000 | Other machines, with built-in centrifugal drier |

**Figure 6:** TARIC search hits on ‘centrifugal’.

CN/TARIC is then queried on each of these significant words, or part of them. For instance, the root word *'**centrifug**'* has four entries in TARIC: *centrifuge*, *centrifuges, centrifugal* and *centrifugation*. For the words *centrifuge* and *centrifugation*, no entry was found and no TARIC item identified. For the two other words, *centrifuges* and *centrifugal*, 4 and 7 entries were found respectively. It should also be mentioned that more than one significant word can be present in the description of a single CN item (e.g. 8421910000 in Figure 5 contains both *centrifuge* and *centrifugal*). 8 different results are therefore reported.

The result for the other significant words are:

- ***rotor*** with two entries (*rotor* and *rotors*) provide 4 and 2 hits respectively (total of 6).
- ***cylinder*** with three entries (*cylinder*, *cylinders*, *cylindrical*) provide 66, 13 and 27 hits respectively (104 in total).
- ***corrosion*** with *corrosion* and *corrosive* provides 5 and 1 hits.
- ***wall thickness*** gives 16 hits all together.

No result was found for *rotor(s) tube(s) assembly(ies)* nor with *gas centrifuge(s)*.

This short example search highlights a few specific points.
First of all, for one given item (in our case 0B001b11) the number of hits in CN-C can reach very high values (134 hits were obtained). Some of the CN-C items found can appear twice under two different entries. If we intend to search over a number of 50-100 different items as explained in section 4, the order of magnitude of hits to be processed will typically reach a few thousands.
A possible way to reduce the number of hits is to filter manually the significant words automatically extracted, either eliminating them from the list, either choosing the multiword expression(s) that include the term. For instance we can replace "resistant" by "materials resistant" or instead of searching for "cylinder" we can search for "rigid cylinder" and "tube cylinder". This will reduce the recall but could influence as well the precision of retrieval.

## 6. Discussion

**Classification schemes.** The classification used in the INFCIRC 254, as well as the one used in the Council Regulation on dual-use follow a logic very close to the description of the nuclear fuel cycle. Items go along a line of fuel cycle: enrichment (0B001), transfer of UF6 (0B002), conversion (0B003), Heavy Water production (0B004), fuel fabrication (0B005), reprocessing (0B006), Plutonium conversion (0B007), etc. In each of these chapters, a few sensitive items are selected and characterized.
From the customs point of view, the goods' classification system used in CN/TARIC is a type of 'partition' of the space of goods that are subject to trade. This means that, a given item (whether dual-use or not) is classified in only one way in a tree-like system. Typically, specialists in goods nomenclature establish a categorisation in a top-down fashion.
On the other hand, an exporter has to face the inverse problem: which CN-C value to assign to a given item in order to fit with the spirit of the classification system?
A further issue to keep in mind is that the 'words of significance' in the nuclear dual-use and in the CN system do not always match due to a different culture in the choice of terms.

**Making the best use of results.** For one given EU-DU item, the search over the significant words provides up to a few hundreds CN-C items. If we intend to search over 50-100 priority items as explained in section 4, the order of magnitude of hits to be verified will typically reach a few thousands. In order to get to the CN-C of interest, one has to eliminate all the irrelevant ones.
One way of helping this is to restrict the search to those CN chapters of interest. For example, some chapters can be excluded, such as 'Live animals; animal products' (chapter 1 to 5). In a first trial we have reduced the number of chapters to be searched by 39 chapters over a total of 97.
Another way to eliminate the irrelevant CN codes is to check one by one the CN items detected against the DU definition. This means that each given EU-DU item will be checked hundreds of time against each of the hit CN-C, until all the unnecessary CN-C are eliminated. The result is a list of possible CN-C for the given DU item.
A totally different approach would be to tackle the EU-DU classification problem by revising and embedding in the Combined Nomenclature ad-hoc CN-C codes, codes that refer to DU items in a non ambiguous way ideally leading to a clean one-to-one correspondence between EU-DU-C and CN-C.

**Use of languages: translations.** Another issue is the place and the role of translations. The translation of texts (EU-DU-C or CN-C) is not an exact science: a given word can be translated in more than one way depending on its context. Then, matching words can introduce some approximation or be incomplete, whilst the sense will continuously be matched.
For example, the word *diameter* (English) hits 145 items in the English version of TARIC, the word *diamètre* (French) hits 151 items in its French version and *diametro* hits 155 items in the Italian version.
This may decrease the probability of catching the 'right' CN-C items for a given DU item and it also makes the results dependent on the working language in which the search is performed.

**Synonyms.** Further, one should take into account the existence of synonyms. The words *tube* and *pipe*, *cylinder* or *tank* can refer to the same reality. This analysis can be assisted in an automatic way by the use of dictionaries of synonyms -preferably made by nuclear specialists and not general ones. Taking synonyms into account is expected to increase the number of hits (the 'recall') although one can hope to increase the retrieval precision as well.
In the end, we feel that the evaluation of the results by a domain expert will still be needed for deciding between relevant and irrelevant items.

**Steps ahead ?** Language technology has proven to be effective to filter and select 'free texts' that are originated in the open source. On the other hand, the CN definitions and their very structured organization cannot be ignored when trying to match DU descriptions. Working upstream, together with CN specialists can be a way to explore.

## 7. References

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# Proliferation Sensitivity of Dual Use Equipment for Laser Isotope Separation

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***Abstract:***

*The international nuclear security is strengthened by explicit import/export control of dual-use equipment. This paper reports on a methodology applying Fault Tree Analysis (FTA) to enhance this control by detecting changes in import behaviour. The methodology is applied to the import of a combination of components that might be used for the construction of Laser Isotope Separation (LIS) plants.*

*The critical components that are necessary for the construction of a LIS plant are systematically described in a Tree structure. The Tree is analysed by the Fault Tree Analyser ASTRA developed in the safety domain. The fault tree can be analysed qualitatively or quantitatively. The qualitative analysis consists in determining the Minimal Cut Sets (MCS) and the Structural Components' Importance Indexes. In our application an MCS represents the minimum number of components categories needed to install a LIS plant. The quantitative analysis is possible when probabilities can be associated with the components. In our case the probability of a component is linked with the probability of importing the corresponding components class. As a first tentative the probability has been defined as the ratios of the financial value of critical components' export to one country to the financial value of the total export to all countries*

*The evolution in time of the Top Event probability is obtained by analyzing real data from five year periods, in seven subsequent time frames, starting with the period 1995-1999 towards the final period 2001-2005. This technique demonstrates the monitoring potential based on import statistics changes in behaviour of a country with regard to imported LIS components.*

*Advanced versions of this probabilistic method may provide customs services with a reduced watch dog list of sensitive components of LIS for a given country. It might also provide an effective tool that is of potential use for registering the capability of a country to setup LIS and for monitoring changes in import behaviour of this country with regard to LIS components.*

**Keywords:** Laser Isotope Separation (AVLIS, MLIS), dual use, enrichment, Fault Tree (ASTRA), Combined Nomenclature (CN) Codes, non-proliferation

## 1. Introduction

Explicit import/export control of the equipment in the dual-use list (INFCIRC/254 part II) strengthens international security. However, in the field of Laser Isotope Separation (LIS) it is a challenge to detect nuclear technology transfer because of the small size of such plants and the steadily upgrading components [1, 2, 3]. Laser enrichment techniques are still under research [4, 5, 6, 7, 8] and as such the components under development have not made their way to the dual use list yet, even though the latter is periodically amended.

Many countries, that had initiated a LIS programme assessed that the Laser Isotope Separation (LIS) technique suffers from very low throughput and is therefore not competitive. The Separation of

Isotopes by Laser Excitation (SILEX) however has proven commercial viability. It is clear that the recent technical progress enhances the risk of successful laser enrichment programmes, and so the need for strengthened control.

LIS techniques (Atomic Vapour LIS (AVLIS), Molecular Obliteration LIS (MOLIS), Chemical Reaction by Isotope Selective Laser Activation (CRISLA) and Separation of Isotopes by Laser Assisted Retardation of Condensation (SILARC)) have separation factors that are considerably higher than current industrial techniques and are more energy efficient [9, 10]. A small number of separation units (a factor $10^2$ smaller than in the case of centrifuges) is required to achieve high separation factors and therefore many traditional detection techniques are less effective for laser enrichment plants.

Current developments in up-scaling the technique seem promising. An industrial application of this technique might be a fact in the near future. Additional verification techniques, such as this study, that focus on the appropriate technical specifications of the different components could help in enhancing the estimation of a country's capability to construct a LIS plant.

## 2. Problem Statement

This work applied a probabilistic risk assessment approach to assess the possibility of establishing laser enrichment activities. The final goal is to develop a system that can give an early indication for detecting changes (in particular increases) in potential capabilities of establishing LIS plants. Given are import/export statistics for well-known countries (i, i+1, … i+k) to a country subject of our analysis with unknown production capacities regarding components useful for LIS. Figure 1 sketches the problem for a subject country X. The solid line arrows towards country X describe the import of components, of which some might be useful or specific for LIS. The dashed line arrows describe the bilateral relation back with the export of components from country X. The weight of the arrows represents the financial volume of the total transfer of components. Some special cases are:

- absence of solid line arrows: country X is impeded to import (e.g. embargo)
- absence of dashed line arrows: country X has no national production capacity developed for export
- the weight of the solid lines equals the weight of the dashed lines: country X is a transit country

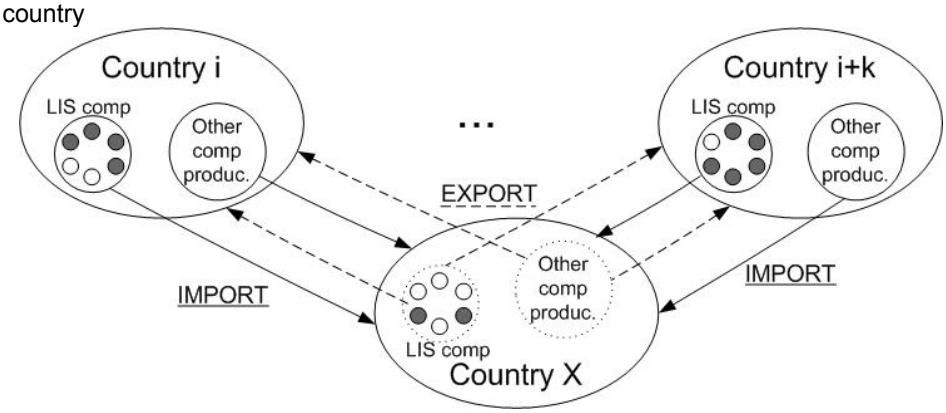


**Figure 1**: Schematics of the problem to extract useful data on LIS capability for a country based on import/export data.

The underlying assumption is that if the imported components are such that they form almost a complete set to construct a LIS, this import is with reasonable probability an indication of the will of the country to construct a LIS plant. However, the strong suspicion on potential LIS development needs then to be confirmed by the results of further investigations, with open source data, on e.g. the country's nuclear technology and uranium reserves, or data from illicit trafficking databases.

It is left up to the user (the inspector) to introduce a threshold of evidence above which an alarm is triggered. This threshold should represent the percentage of components present in the country to the total percentage of components needed to construct a LIS. (An example is represented in figure 1 by filling dark only the present components out of all necessary components.) Such threshold should take into account the minimal required size of a laser facility for isotope enrichment.

## 3. The Tree of the Laser Isotope Separation System

The Tree system of figure 2 has been constructed such that it incorporates the two laser techniques: Atomic Vapour LIS (AVLIS) and the Molecular LIS (MLIS, including MOLIS, CRISLA, SILARC). The necessary components that are required to construct a AVLIS or MLIS are determined from open literature. The AVLIS and MLIS can be developed in parallel and therefore are not mutually exclusive systems.

Both need a laser system with analysis instruments and optical equipment (A1). For the AVLIS option, an evaporation and collection system is operating under an electromagnetic field and at vacuum and hence all components of GAVLISDET are needed. For the MLIS options the slightly heated $UF_6$ gas needs to be compressed through a supersonic nozzle that is then cooled, hence all components of the GMLISDET are needed. The subdivision of a system in its components was repeated until the basic components are part of the internationally defined categories of components that are characterised with a Combined Nomenclature (CN) code index and used by the Customs in the EU. There are 21 basic events, named with the CN code.

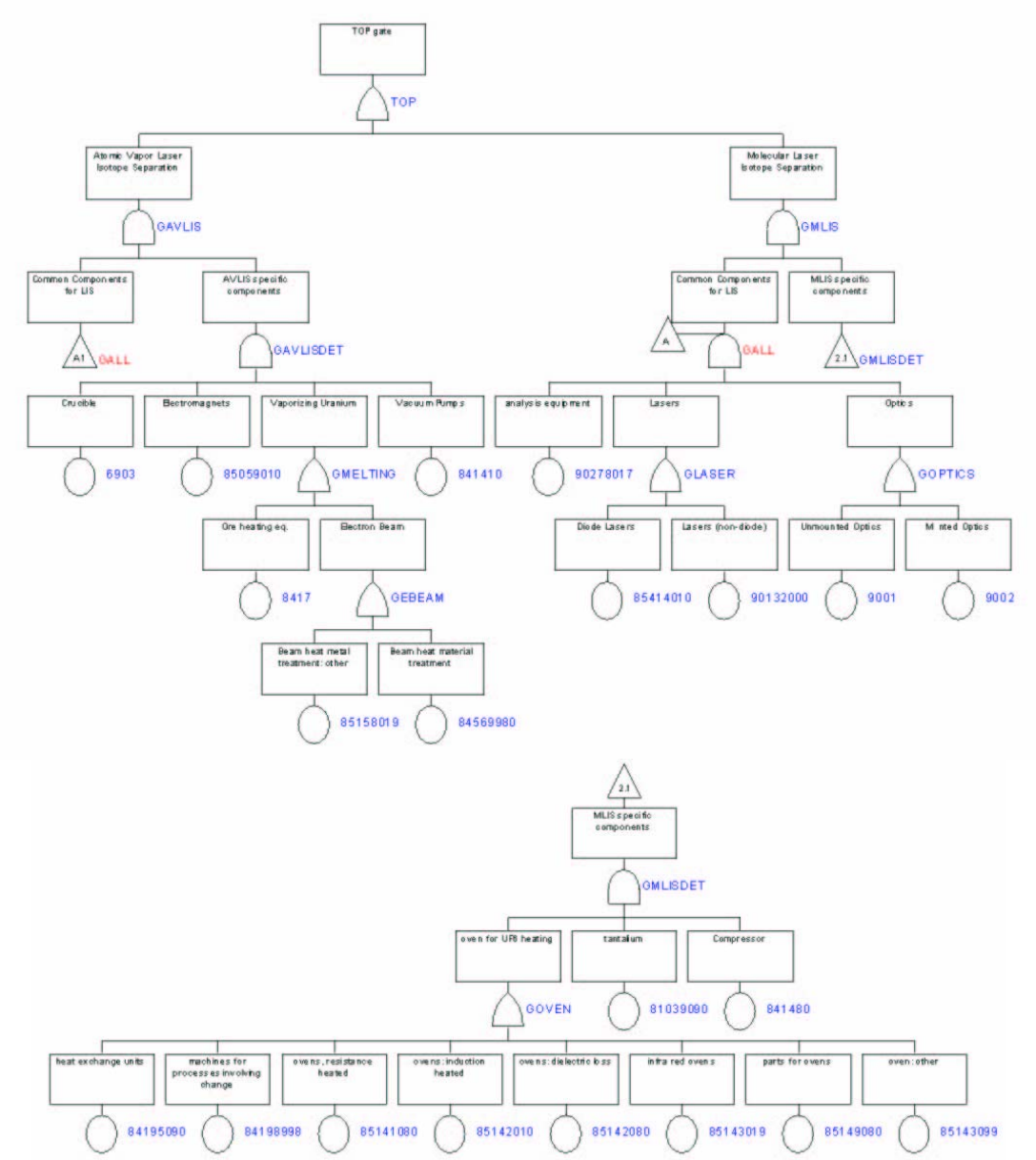


**Figure 2:** Fault Tree used for analysis

The analysis of the Tree determines the Minimal Cut Sets, defined as the smallest number of components necessary for the construction of a LIS plant. The tree in Figure contains 44 MCS, 12 of sixth order and 32 of seventh order. The structural importance of a component is related to the number and order of MCS in which it is contained.
The component category with highest structural importance was the analysis instruments for the laser system (CN 90278017). The list of the 15 component categories ranked in order of diminishing structural importance is given in Table 1. With this additional knowledge, the current list of dual use items might be upgraded with a ranking system for the most sensitive components common to LIS systems. It could also lead to a reduced watch dog list for a certain technique in a given country.

| ***CN Code of basic event*** | ***brief description*** | ***structural importance[1]*** |
|---|---|---|
| 90278017 | Physical and chemical analysis instrumentation | 18.63 % |
| 81039090<br>841480 | Articles of tantalum<br>Air compressors of various types | 12.47 % |
| 9001<br>9002<br>85414010<br>90132000 | Unmounted optical equipment (lenses, prisms, mirrors, etc.)<br>Mounted optical elements (part of apparatus, fitting)<br>Light emitting diodes<br>Laser (excl. laser diodes) | 6.21 % |
| 6903<br>841410<br>85059010 | Retorts, crucibles, nozzles, tubes, other refract. ceramic goods<br>Vacuum pumps, diffusion pumps, cryo-pumps, adsorption pumps<br>Electromagnets (not for medical use) | 4.62 % |
| 8417<br>84569980<br>85158019 | Parts of electric industrial and lab furnaces and ovens<br>Material removing machines (electron/ion beam, plasma arc)<br>Electrical machine for hot spraying of metals or metal carbides | 0.66 % |
| 8514 | Furnaces & ovens:electrical resistance, induction, dielectric loss, infrared | 0.29 % |
| 84195090<br>84198998 | Heat exchanging units<br>Device for treating materials (process based on temperature gradient) | 0.05 % |

Table 1: component categories with structural importance (The last column indicates the total contribution to Minimal Cut Sets in percentage)

## 4. Evaluation of Top Event Probability

### 4.1. Coupling of Import/export Statistical Data to the LIS Tree Components

To monitor a country's change in infrastructure and equipment with regard to laser enrichment requires a careful analysis of import/export statistics for that country and a good estimation of the internal production. For each imported category of components the total financial volume, as reported by the Customs in the import/export statistics is available. In this first approach neither the export statistics of the subject country nor the national production is evaluated. No number of components is available for the different categories, and so a direct relation for the probability of imported components is lacking.

In the example underneath the probability value for each imported dual use component is approached by the export fraction of a component for the LIS towards a subject country to the export of that component to all countries that are present in the database. More in particular, the cumulated export value of each LIS component from a predefined group of countries to this country over a certain period of time was divided by the sum of export of the subject component to all countries in this time period. More details can be found in [11].

### 4.2. Methodology Applying Fault Tree Analysis

[1] Total contribution in percentage

The example underneath is worked out using the Fault Tree Analysis (FTA) Technique. The Advanced Software Tool for Reliability Analysis (ASTRA), developed at the Joint Research Centre (JRC) of the European Commission [12] is used to analyse the Fault Tree. The analysis returns a value that can be interpreted as an indication that a state could be developing laser enrichment program with the imported components. The "Top Event" is defined as "the development of a laser enrichment programme with imported dual use components". The underlying events are not characterising "fault of a component" but "import of the dual-use component". Combined in the tree the probability to successfully acquire the necessary components is represented.

The ASTRA analyser yields the Top Event probability and its evolution in time due to the yearly changes in import of the dual-use components. These changes in import behaviour of the subject country compared to the other countries are then monitored in time with the cumulative distribution of all imported components. Biases that are caused by e.g. an economic crisis or an embargo become visible in these plots.

This approach is then used to register the relevant changes in capabilities of countries to construct a LIS plant. It is found that more significant results are obtained if the time periods are comparable with the duration of an economic cycle (5yr) and overlapping. The first analysis starts with data from the period 1995-1999 and subsequent five year periods are analyzed. Since this study is based on export data available from 1995 onwards the analysis was limited to seven time windows.

Obviously the availability of expertise in many scientific and engineering subjects is necessary to set up a LIS plant. Since the spread of knowledge and expertise cannot be measured using export data, it is not included in this first approach. One way of implementing such information is to monitor the scientific output in the subject area of the country's research centres, if it is published. The use of Expert Opinion based on open source information can be applied to define the probabilities of present knowledge and expertise.

### 4.3. Example of Monitoring Import Behaviour for Different Scenario's

To perform a quantitative analysis for different countries, three families of countries have been distinguished:

- The O family describes industrial well-developed countries with stable import/export. Countries with a Human Development Index (HDI) > 0.75 are allocated to this category.
- The A family describes a country of rapid economic growth, to which countries with $0.50 < HDI \leq 0.75$ are allocated.
- The B family describes very heterogeneous countries with slow economic growth, composed of countries with $HDI \leq 0.50$.

Based upon those families two storylines or scenarios are considered:

- The AB storyline applies to a country of family A with growing nuclear fuel cycle that could easily transform this cycle to a military one when surrounding conditions let it evolve to family B.
- The BA storyline applies to a country of family B with nuclear research an activity that abstains from its original military intentions, when evolving to family A.

Disposing of the export statistics for most CN codes from European to non-European countries a series of non-European countries have been selected to investigate their import behaviour in view of the above mentioned scenarios. These importing countries were divided into the above defined families O, A and B. Figure 3 presents the typical import behaviour for countries of each family. A relatively constant import behaviour over the short time period is clearly present. The order of the probability for the Top Event is very low due to the used financial volumes for the determination of the probability of basic imported components. Moreover this probability is closely related to the level of industrialisation and trade relations of the importing countries with the European exporting countries, which is for rather small countries with limited trade expected to be orders of magnitude lower compared to large countries with well-established economy. Other definitions of probabilities more appropriate for this type of analysis are under study.

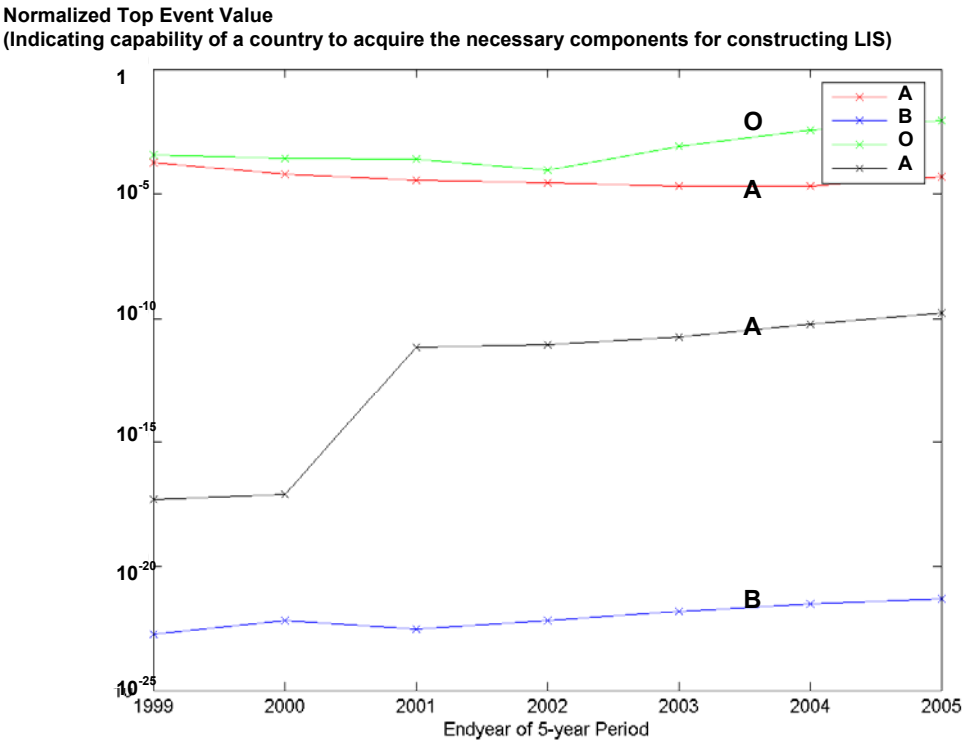


**Figure 3:** Monitoring of Import Behaviour (Normalised Top Event Value) over five year periods for countries of family A (Medium human development index (HDI)), B (low HDI), and O (high HDI)

For family A, the analysis was extended to a second country to illustrate the effect of significant increase in import behaviour, which was expected to have developed over the history a potential scenario. For that case a rising trend from 2001 onwards is clearly visible. More analysis of countries adhering to scenarios for countries of family A are performed, refining the group of countries. In addition one year time periods were applied to make abstraction of the existing infrastructures or equipment. These analyses show consistently a peak in 2001, which is supposed to be caused by a shift in economical dominance due to an economical crisis. This example illustrates also that the assumption of scenario AB should be used with precaution.

### 4.4 Criticality of imported components for a monitored country

The criticality index of a certain component expresses the relative variation of the Top Event probability caused by the relative variation of this component's probability. In the example here the Top Event probability refers to a financial budget with which components for potential use in LIS plants can be bought, which changes with the market prices. An actualisation of the total volume needed to buy the components of a LIS plant is needed to monitor the criticality index of a certain component in time.

Each variation in import behaviour of a country (as show in figure 3 for some countries) needs to be combined with an allocation of the change in imported components. In particular the typology of the imported components over time needs to be monitored, what is planned in a further study.

## 5. Conclusions and Perspective

It was proven beneficial to setup for a given proliferation sensitive system a Tree and to derive the minimal cut sets. This provides an identification of all components with structural importance for LIS construction. Depending on the ratio of present components in a subject country to the total number of necessary components a threshold could be set, that activates a flag of suspicion and launches additional investigations. With the components of structural importance the current list of dual use

items could be upgraded with a ranking system for the most sensitive components common to LIS systems.

In a first example data from import/export statistics was coupled to the Tree and the Top Event was evaluated with the Fault Tree Analysis Technique. It was demonstrated that the method has the potential to indicate anomalies in import behaviour of dual use components of any kind for a given subject country. More info, in particular the number of imported components is needed to establish a more direct relation to the import/export data and the probabilities needed for the Tree.

Advanced versions of this probabilistic method may provide customs services with an effective tool that is of potential use for detecting anomalies in import behaviour of LIS or other sensitive technologies. Future research might compare the proposed FTA technique with alternatives such as Bayesian Networks.

## 6. Acknowledgements

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# Assessing the confidence the Agency has in the broader conclusion : the role of a quality management system in supporting this assessment

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***Abstract:***

*In drawing the broader safeguards conclusion the Agency evaluates all of the information available to it under comprehensive safeguards agreements and the additional protocol. The vast amount of information now available to the Agency through State declarations, in-field verification activities and open and other sources of information cannot be systematically verified, but can only be assessed on a selective basis. However, even though all the information is not systematically verified, the verification that is done must be carried out in a systematic way and the results fed into the State evaluation process. The need for a formalized management system to guide this work through a formal plan, do, check, act cycle is paramount if the Agency is to continue to deliver soundly-based safeguards conclusions to the Board of Governors. This system must ensure that the analysis work is carried out systematically but not stifle the flair and creative thinking necessary of the analysts doing the work. It will also encompass the management of the Agency's knowledge assets and will bring other tangible benefits, such as the ability to respond quickly to a changing safeguards environment by managing that change more effectively.*

*The paper describes the value that a formal management system can bring to the work of the Agency. It discusses the problem of assessing the confidence in the broader conclusion that all nuclear material in a State is in peaceful use and, in particular, how the Agency decides that sufficient information has been analysed and having not found - or found and then resolved - any indicators, it can be concluded that enough has been done and the broader conclusion can be drawn. The paper examines how we make sure that we become more watchful for different creative and imaginative proliferation scenarios when historically we have been focussed on classical safeguards techniques based on verification of declared nuclear material and activities.*

**Keywords:** quality, conclusion, system, evaluation

## 1. Introduction

How does an organization assure the quality of its products when they are conclusions regarding the non-diversion of nuclear material or the absence of undeclared activities? Such conclusions cannot be quality controlled in a traditional sense, like manufactured components, and unlike a modern service industry the 'customer' is not able simply to choose another service provider if it perceives that the products are of poor quality, however it may define quality.

This is the kind of challenge that the IAEA has always faced and never more than now, as an increasing number of States with comprehensive safeguards agreements in force also conclude additional protocols. The safeguards measures provided for in additional protocols equip the Agency with important, supplementary measures which enhance its ability to detect undeclared activities. Safeguards conclusions are no longer drawn solely at the facility level, based on assessment against the well-defined Safeguards Criteria, but are drawn at the level of a State as a whole.

Drawing conclusions regarding the absence of undeclared nuclear material and activities is intrinsically more challenging than verifying the absence of diversion of declared nuclear material. The requirement from Member States that this conclusion should be soundly based, and thus credible, requires the Agency to study very carefully how it is derived. The familiar evaluation methods used in traditional safeguards when dealing with declared material are no longer capable of delivering the necessary confidence in the conclusions.

Against the background of a fundamentally changed basis in the way safeguards conclusions are drawn, the Secretariat needs to ensure that it has necessary and sufficient processes in place and that the processes are well understood, carried out correctly by staff with the right skills and provide appropriate feedback loops. All of these are important if the conclusions drawn under strengthened safeguards are to be soundly based.

Part of the approach being taken by the Department of Safeguards to meet this challenge is to implement a quality management system (QMS) based on ISO 9001:2000[1], [2]. This paper describes how implementing a QMS can help give confidence in the broader safeguards conclusion and discusses other approaches that could be used to assess the conclusions in a qualitative manner.

## 2. The dilemma of drawing a conclusion regarding the absence of undeclared nuclear material and activities

Safeguards conclusions are based on evaluation of all the information available to the Agency in exercising its rights and fulfilling its obligations at that point in time.

To conclude that there is no indication of diversion of declared nuclear material from peaceful nuclear activities in a State, the Secretariat carries out a comprehensive evaluation of all information available to it. This encompasses the information provided by the State with regard to the design and operation of declared nuclear facilities, the State's nuclear material accounting reports and the results of the Secretariat's inspections carried out in order to verify the State's declarations. In addition, the Secretariat evaluates the information acquired through the implementation of the State's additional protocol.

To conclude that there is no indication of undeclared nuclear material and activities in a State, the Secretariat carries out an evaluation of the consistency of the State's declared nuclear programme with the results of its verification activities under the relevant safeguards agreement and additional protocol and with all other information available to the Agency. In order to draw this conclusion, the Agency needs to have:

- conducted a comprehensive State evaluation based on all information available to the Agency about the State's nuclear and nuclear-related activities (including declarations submitted under the additional protocol, and information collected by the Agency through its verification activities and from other sources);
- implemented complementary access, as necessary, in accordance with the State's additional protocol; and
- addressed all anomalies, questions and inconsistencies identified in the course of its evaluation and verification activities.

When these evaluations have been completed, and no indication has been found by the Secretariat that, in its judgement, would give rise to a possible proliferation concern, the Secretariat can draw the broader conclusion that all nuclear material in a State has remained in peaceful activities.

The dilemma of drawing the broader safeguards conclusion relates to concluding that there is no indication of undeclared nuclear material and activities in a State. If evidence of undeclared nuclear material and activities in a State is sought, but not found, then two possibilities exist. Either there was something to find but the Secretariat didn't look in the right places, or there really was nothing to find. In drawing the conclusion that all nuclear material in the State remained in peaceful activities, the Secretariat has to be confident that if there had been undeclared nuclear material and activities then the activities conducted would have, with a high degree of confidence, detected the evidence. This is the judgement of what is sufficient. Secondly, if no evidence was found then there were not any

undeclared material or facilities, the declaration was therefore complete and so it can be concluded that all nuclear material remained in peaceful use or was otherwise accounted for.

The framework in which this takes place is the State evaluation process. State evaluations take into account a much wider variety of safeguards relevant information than ever before and reflect the increasing moves away from purely quantitative to more qualitative assessments.

## 3. Why are we implementing a quality management system?

It is not possible to test the broader safeguards conclusion itself, specifically the component that relates to the absence of evidence of undeclared nuclear material and activities. Therefore, to have confidence in the conclusion, we must understand and have confidence in the processes that go into drawing the broader conclusion.

The Secretariat can only be confident in the result if it is able to confirm that the necessary and sufficient processes are in place, they are carried out correctly and adequately monitored and that the appropriate feedback loops are in place. This cannot be achieved by checking the outputs in a classic quality control mode, it is the processes themselves that must be thoroughly understood, operated correctly and monitored. Furthermore the knowledge of what the processes cannot deliver must also be available so that any weaknesses or shortcomings can be handled correctly. In order to achieve this the Department of Safeguards is implementing a quality management system (QMS) based on ISO 9001:2000.

The ISO 9001:2000 Standard was chosen because it is an international system, it is widely accepted and used and, more importantly, it is sufficiently flexible to enable the Department to introduce a system appropriate to its own needs.

The need for a comprehensive QMS was recognised internally by the Department of Safeguards in light of the changing safeguards environment and increasing moves away from quantitative to more qualitative assessments. It was also recognised by a number of other stakeholders such as Member States and the Director General's Standing Advisory Group on Safeguards Implementation (SAGSI). More recently, the need for implementation of Agency-wide quality management practices was reflected, inert alia, in the Agency's Medium Term Strategy for 2006-2011.

## 4. How can we assess the confidence of the broader conclusion?

There are a number of different aspects to assessing the confidence in the broader safeguards conclusion. It is not possible to assign a statistically based and quantifiable confidence level to the broader safeguards conclusion as it is to the conclusion regarding the non-diversion of declared nuclear material. Therefore we have to identify a range of measures that will allow us to assess, more qualitatively, the confidence in the conclusion.

A number of options can be considered. Firstly the Department can test whether the work processes are operating as planned. The QMS helps to ensure that appropriate processes are planned, implemented and continually improved; however it is necessary to look deeper than the operation of the processes to assess the confidence in the conclusions. Secondly, the responsiveness of the system to different types of information can be assessed. Thirdly, the assumptions that are built in to the evaluation process need to be understood and the impact changing an assumption would have on a conclusion. Finally, the roles that people have in the evaluation process and the role of organizational culture must also be considered.

At the process level a number of different components of the QMS allows confirmation that the processes that go into drawing the conclusions are working effectively. For example, the internal quality audit process can be used to check that we are implementing the evaluation process as intended. Secondly continual process improvement can be used to identify ongoing improvements to the State evaluation process and implement changes in a controlled manner. Process mapping also allows the interfaces with other processes in the Department to be clearly defined, transparent and understood by all of the stakeholders.

An alternative type of assessment is to test the responsiveness of the evaluation process to different inputs. This could be conducted by introducing some false-positive data into the process and assessing how these data are dealt with within the system. However there are difficulties and risks to this approach to testing the system. Firstly the introduction of any data would have to be carried out with only a minimum of staff knowing of the "test". Also being able to monitor the progress of the false-positive data through the process would be difficult. There is also a risk of the false-positive information causing an alarm that would need halting before further corrective action was taken, for example, carrying out verification in a State based on the false-positive information. Therefore if this type of approach were to be used, a clearly defined protocol would need to be put in place to ensure no inappropriate corrective action was implemented.

An alternative method of assessing the confidence in the process and the conclusion would be to carry out two evaluations for the same State, but using two independent teams with access to the same information to carry out the evaluation. Any differences in the conclusions drawn between the teams could then be reviewed and corrective actions identified. This leads to a further method of testing the State evaluation process – understanding what assumptions are made during the analysis and evaluation of information and what impact they have on the safeguards conclusion.

Understanding the assumptions that underpin the evaluation process is important because they can be a potential source of bias in the process, which would not necessarily be identified by other methods such as internal quality auditing [3]. The analysis of assumptions and the impact they have on the evaluation process can be viewed from a preventive action perspective. The use of flawed or incorrect assumptions is a risk to the process operating effectively and appropriate preventive measures should be implemented to reduce the consequence and likelihood of the risk occurring.

The assumptions can range from ones that work at an organizational level to those based at an individual level. Three possible assumptions are described below as examples. They are defined in a provocative manner to illustrate the importance assumptions can play in evaluation and the need for the Department to understand what impact they could have.

Assumption 1 Drawing the broader safeguards conclusion about a State is a good thing to do and should be encouraged, as it will move the State to integrated safeguards and thus provide more effective and efficient safeguards.

If this assumption is true what are the consequences? Does it bias the process from the outset so that information supporting the broader conclusion is preferentially used in the State evaluation process?

The alternative assumption is that the Secretariat must make sure that the correct conclusion is drawn on a sound and non-discriminatory basis, regardless of any perceived need to draw the broader conclusion and move to integrated safeguards.

Assumption 2 Do we assume the State is telling the truth in its declaration, it is trying to hide something or do we assume it is up to the Secretariat to verify the declaration and other information as necessary?

Assumption 3 The Department of Safeguards is historically an organization that verifies declared nuclear material and activities. Has this history created an organizational culture focused on verifying declared activities and nuclear material and has difficulty in effectively undertaking "what-if" analysis to assess alternative and creative diversion scenarios and clandestine activities?

The purpose of describing these assumptions is to highlight the importance of understanding what assumptions are within the State evaluation process, assessing how they can be tested and making sure measures are in place so that safeguards conclusions are drawn on a sound basis. This work must be considered within the context of process design to ensure that the conclusions drawn are free from bias hidden within any assumption.

The final area to consider in assessing the confidence of the broader conclusion is the impact that people have on the State evaluation process. At the basic level this means that staff have the correct skills, experiences and competencies to undertake the evaluation process and these are defined within the framework of the QMS. As in most processes, the role of people introduces the most variability, however it also allows for the creativity and analysis to be able to deal with a wide range of diverse data and identify inconsistencies and indicators of possible undeclared activities. Therefore when teams are put together to undertake a State evaluation they are brought together from a range of disciplines within the Department and Secretariat that ensures a conclusion is considered from a range of perspectives, before it is independently peer reviewed by an inter-Department Committee.

The discussion above represents a range of options that could be used to help assess the confidence in the broader conclusion. Each option tackles the issue from a different perspective that when combined, allows for greater confidence in the final result.

## 5. Conclusions

Drawing conclusions regarding the absence of undeclared nuclear material and activities is intrinsically more challenging than verifying the absence of diversion of declared nuclear material. A wide range of new tools are available to the Agency to contribute to this process. However the Secretariat can only have confidence in the soundness of its conclusions and thus provide credible assurances to the international community that States are complying with their safeguards obligations if the necessary and sufficient processes are in place, they are carried out correctly and adequately monitored and that appropriate feedback loops are used. It is simply not possible to carry out quality control activities to verify the absence of undeclared nuclear material and activities.

The implementation of a quality management system within the Department of Safeguards will ensure that it has necessary and sufficient processes in place, that the processes are well understood, carried out correctly by staff with the right skills and competencies and provide appropriate feedback loops. However the QMS must also be used to help the Department understand not only how the tangible inputs such as results of all verification activities, in field and at headquarters, analysis of State declared information and review of open source information combine to draw a conclusion, but also the less readily identifiable inputs such as understanding the assumptions that underpin the State evaluation process.

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# Session 26

# Communication, Education & Training

# The safeguards discourse as a pathway for transformations in public opinion

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***Abstract***

*The starting point of this paper is based on the principle that the pro-nuclear discourse is not as strong as the anti-nuclear discourse in sustaining its arguments. Even decades after the Second World War, the accidents with a great repercussion and the numberless attempts made since then to regain society in favor of nuclear activities, this discourse does not appear to be in a condition to face its opponents successfully because what is sustained in its wording does not work as well as it should as an instrument for persuasion and/or clarification of public opinion. No matter how paradoxical it may appear, the discourse in favor of nuclear technology has also been of use for prompting negative attitudes in this regard. In turn, such attitudes interfere with the perception of the nuclear risk and, consequently, with the decision-making process concerning issues such as, for instance, the installation and/or permanence of nuclear power plants in certain countries —regardless the different realities in each one of them. To the extent in which, within this sector, the contact with the population becomes more active among the institutions developing nuclear technology in all its possible peaceful applications, would nuclear safeguards have any discursive influence upon the behavior of society should the latter become involved in the issue of the stigma? Would they be liable to aid towards a more favorable public opinion in connection with this topic? In short, this article attempts to evidence the reasons for this discursive weakness —using the Brazilian case as an example— and to discuss whether nuclear safeguards can play a worth-practicing role in this process.*

**Keywords**: pro-nuclear discourse, stigma, public opinion transformation, the role of safeguards in society, management of knowledge concerning the nuclear sector.

## 1. Why aren't its goals fulfilled?

It can be proved that, even with a significant re-orientation occurred during the last five years, the discourse in favor of nuclear technology in Brazil involves an anti-nuclear content in its statements because it is captured by a field of significances that, somehow, reproduces the stigma and weakens its arguments. Considering that all those involved —that is, the nuclear institutions, the anti-nuclear activists and society as a whole— are living within the same historical context, a negative reference to such technology is practically unconscious, even when it is being praised.

The perception indicating that the discourse by the institutions in this sector is conditioned by the same reasons used by those opposing it leads us to believe that this is the fact contributing to maintain the stigma, making the statements made by these organizations inefficient in attaining their objectives. The polemics continues to be a constant when referred to nuclear issues, whether inside or outside Brazil. This situation does not change because the idea of danger is present in both the attack and the defense discourses, even if implicitly. It appears as if there were a certain self-defense tradition among nuclear institutions. And an attacking tradition among those opposing it — once history demonstrated that danger does exist.

An analysis of some anti-nuclear texts will probably lead to find several statements to which pro-nuclear texts respond in an atemporal manner and vice versa. This situation can be illustrated using, as an example, texts by popular movements for environmental preservation, those by certain non-governmental organizations, some private initiative sites, as well as some journalistic texts, all of them contrary to the use of nuclear technology and, especially, for power generation. At the same time, an

analysis of “pro” texts, we would find ideas —and even expressions— very similar to those "against", thus corroborating what has been said about the fact of a positioning grounded on a given context.

> *Among all industrial activities, power generation in nuclear plants is among the ones involving lesser risk. The dominating criterion is that, in a zero-tolerance environment, safety can always be improved. In over twenty years of nuclear power generation in Angra, there was not a single accident or event involving risks for the plant workers, the population or the local environment. (Electronuclear’s website, 2007)*

> *[...] Leaks or explosions in the reactors due to failures in their safety systems cause severe nuclear accidents. The first one of them, at the Russian plant of Tcheliabinski, in September 1957, contaminated almost 270 thousand people. The most serious one, in Chernobyl, Ukraine, in 1986, caused more than thirty casualties, hundreds of injured and produced a radioactive cloud that spread all over Europe. The number of contaminated people is inestimable.* ***In Brazil, a leak at the Angra I Plant, in Rio de Janeiro, contaminated two technicians.*** *However, the worst accident with radioactive substances occurred in the country was in Goiânia in 1987: the* Instituto Goiano de Radioterapia *abandoned a capsule containing the cessium-137 isotope used in a radiological equipment unit. It was found and open by scrap merchants and, in a short time, killed four people and contaminated two hundred [...]. (BBC Brasil.com – August 5, 2005)*

> *Should everything go wrong… we still have our emergency plans. The seriousness applied by the nuclear power plants in dealing with the safety issue can still be perceived by the so-called emergency plans. Since the time in which the Three-Mile-Island accident occurred, when the plant was not prepared to face an accident of that magnitude, the nuclear sector has been conscious of the need to be perfecting these plans all the time. And, since then, a great evolution can be noticed.* (*Brasil Nuclear* Magazine, April/June 2000)

> *Failures caused an unprecedented incident and raise doubts concerning the safety of the nuclear power plant. […] Nevertheless, the technicians did not celebrate. Locating and correcting the defect was a must. Nervous, they pushed controls, reviewed the operations, scanned the conductors – and nothing. While this happened, the second automatic safety system went into operation in order to avoid the overflow of the tank, which was receiving too much liquid [...]. A leak of this magnitude did never occur in the 16 years of history in the Brazilian nuclear power plants.* (*Epoca* Magazine, September 24, 2001)

When we refer to polemics on this issue, first of all, we want to use, as a starting point, the term leading to debate, as a way to reach the set of significances related to the word. We take up the term as a reference, as a criterion in order to identify the problem. Here, the lexical term [1] is a demonstrative unit, an item that, located within a given context, leads to a certain positioning by means of values that, historically, have been attributed to it.

The reproduction of the nuclear stigma occurs both at the interpretation level by the target audience as at that of the textual production made by the institutions of the nuclear sector. This is due to the fact that the preparation and the consumption of the texts are conditioned, among other factors, by a mechanism of imaginary forms and assumptions that is predominantly negative. Consequently, the audience receives the pro-nuclear discourse with distrust because it reminds people of images connected with the tragedies that surround this sector. In turn, the nuclear institutions release what is antinuclear in their statements because they are affected by the negative images attached by society to this issue and do only take care of denying what is stated in the opposition’s discourse, while leaving untouched the assumptions grounded on the nuclear-equal-danger relationship.

Even when the success of its discourse is guaranteed, the nuclear sector needs to assume what must be said in order to attempt obtaining the desired effect in its target audience. However, all the discursive efforts aimed at guaranteeing safety and the peaceful use of its activities are frustrated when it is found that the public has a negative reaction toward such attempt. This suggests that the assumptions developed for outlining this discourse seem to be wrong, because the discourse is not strong enough to convince the target audience of what it sustains.

In this paper, it is sustained that the main reason for the lack of success of the Brazilian pro-nuclear discourse is a set of negative pre-assumptions concerning what the target audience expects from it. There is a lack of coherence between its intentions and the type of discourse released with regard to its target audience. The discourse elaboration process is filled up with images "soaked" by the anti-nuclear discourse, thus weakening the arguments in favor of this technology. In turn, the target audience discredits such statements due to a lack of confidence and, usually, the media nurtures the reasons for such distrust.

In the following diagrams, keywords repeated in both the pro-nuclear and the anti-nuclear discourses used in the media have been transcribed. The objective is demonstrating that the ideas present in the texts used by the media are also present in the pro-nuclear discourse and can be observed in the references to these ideas made in their statements and in the repetition of several lexical items found in both discourses. There, one can perceive that the grounds for such arguments seem to be in the images proposed at the end of each diagram.

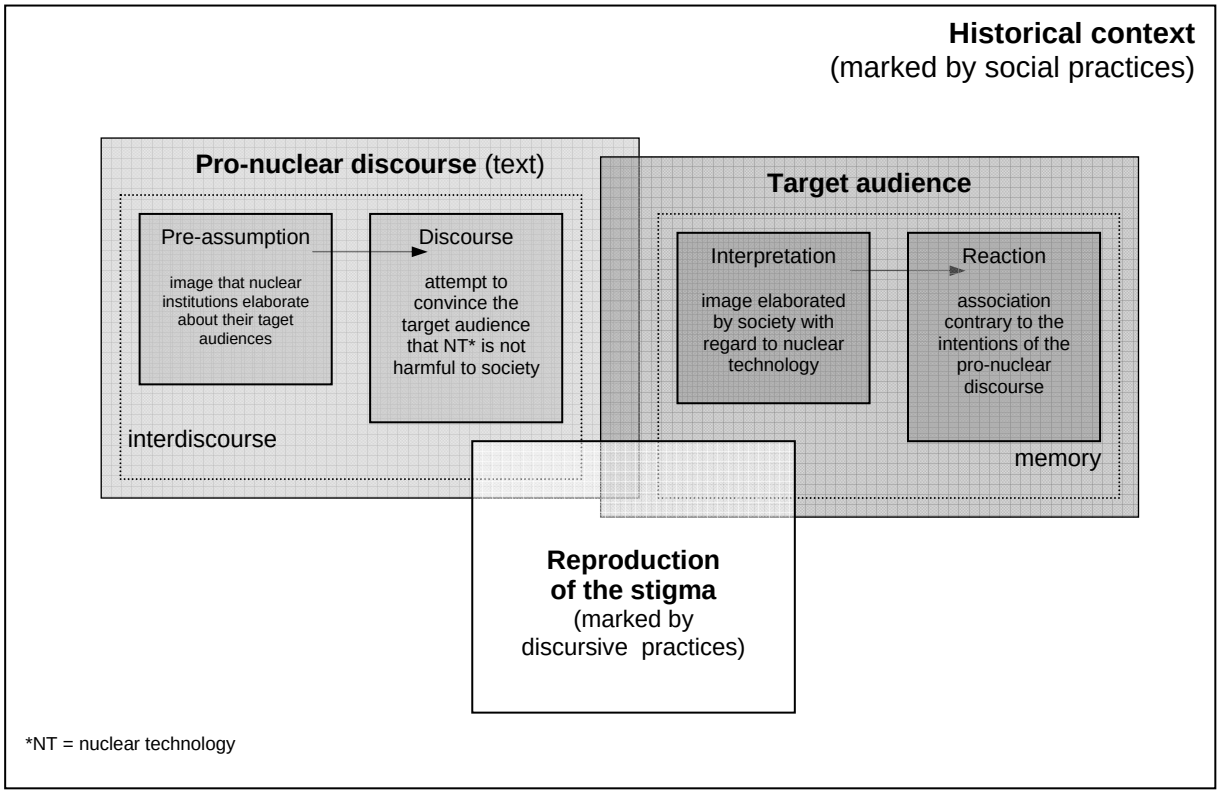
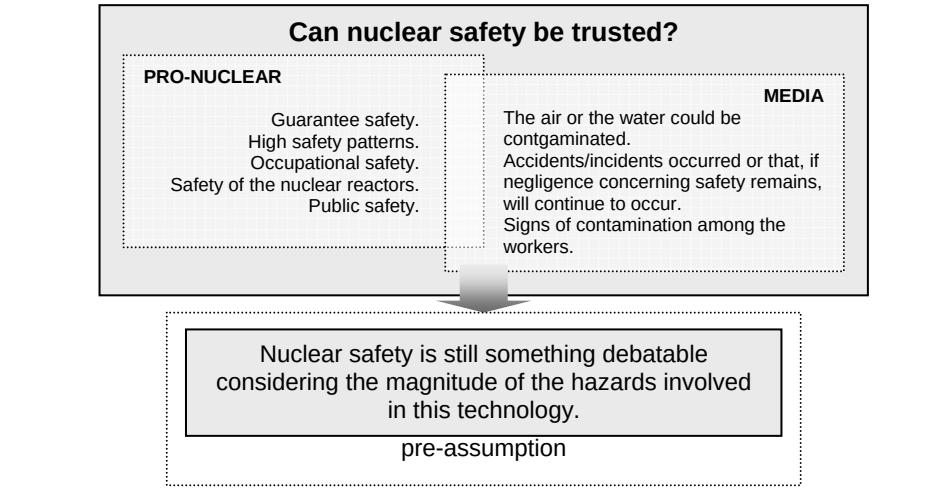
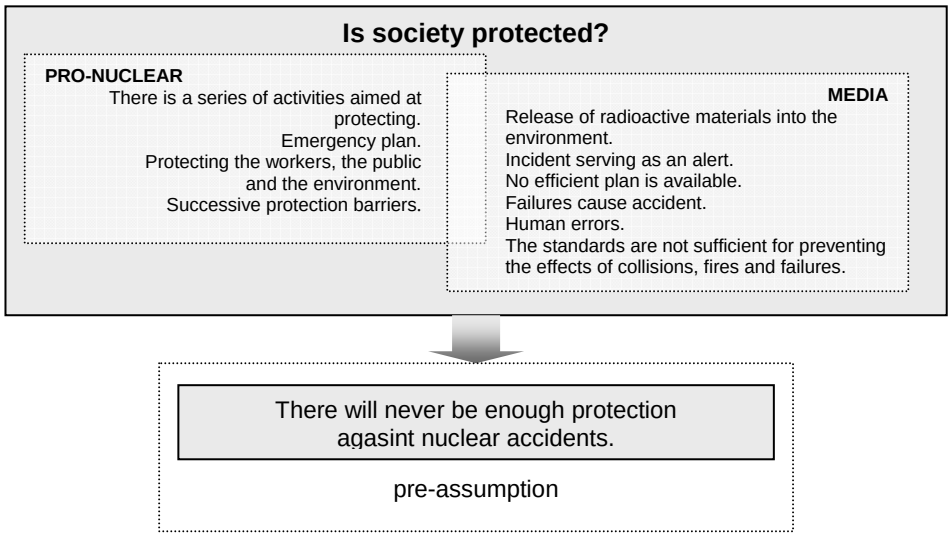
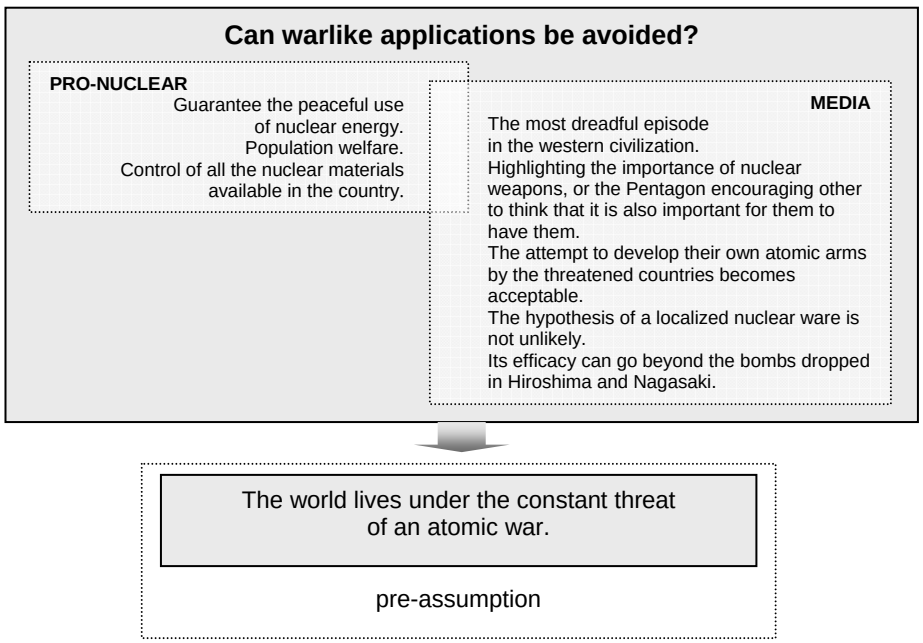
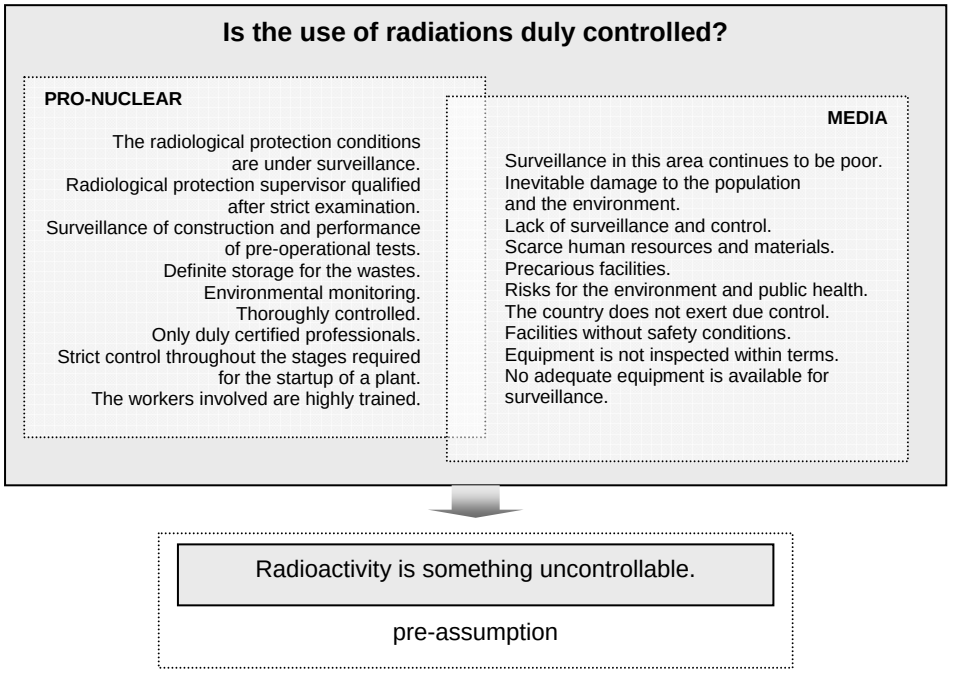


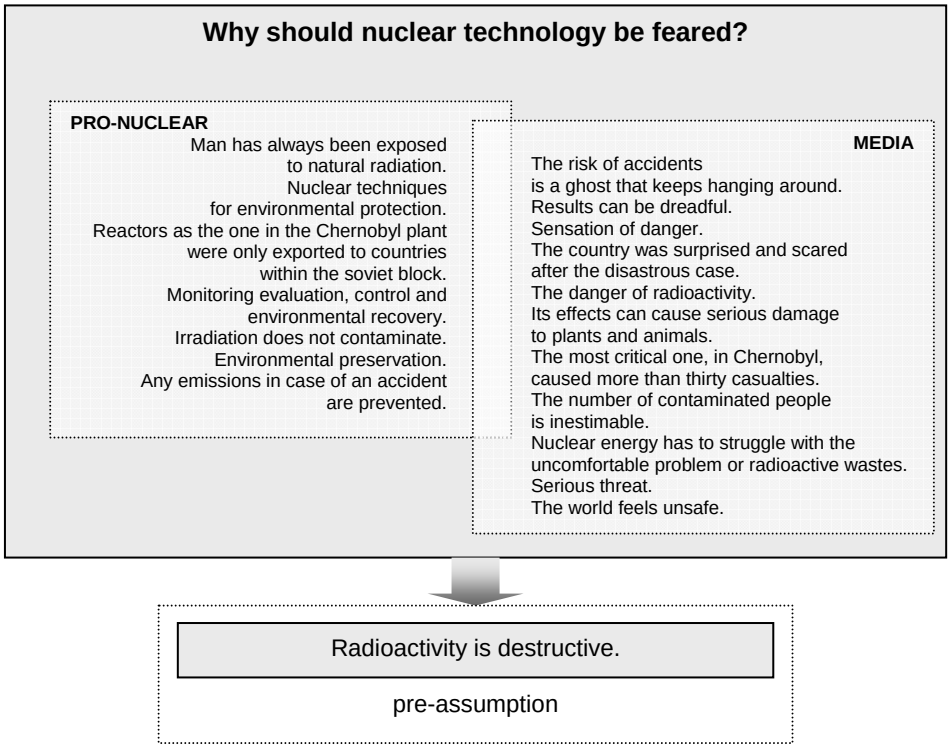

**Fig. 1:** discursive process of the nuclear sector


Can nuclear safety be trusted?
PRO-NUCLEAR
Guarantee safety.
High safety patterns.
Occupational safety.
Safety of the nuclear reactors.
Public safety.
MEDIA
The air or the water could be contgaminated.
Accidents/incidents occurred or that, if negligence concerning safety remains, will continue to occur.
Signs of contamination among the workers.
Nuclear safety is still something debatable considering the magnitude of the hazards involved in this technology.
pre-assumption


Is society protected?
PRO-NUCLEAR
There is a series of activities aimed at protecting.
Emergency plan.
Protecting the workers, the public and the environment.
Successive protection barriers.
MEDIA
Release of radioactive materials into the environment.
Incident serving as an alert.
No efficient plan is available.
Failures cause accident.
Human errors.
The standards are not sufficient for preventing the effects of collisions, fires and failures.
There will never be enough protection agasint nuclear accidents.
pre-assumption


Can warlike applications be avoided?
PRO-NUCLEAR
Guarantee the peaceful use of nuclear energy.
Population welfare.
Control of all the nuclear materials available in the country.
MEDIA
The most dreadful episode in the western civilization.
Highlighting the importance of nuclear weapons, or the Pentagon encouraging other to think that it is also important for them to have them.
The attempt to develop their own atomic arms by the threatened countries becomes acceptable.
The hypothesis of a localized nuclear ware is not unlikely.
Its efficacy can go beyond the bombs dropped in Hiroshima and Nagasaki.
The world lives under the constant threat of an atomic war.
pre-assumption


Is the use of radiations duly controlled?
PRO-NUCLEAR
The radiological protection conditions are under surveillance.
Radiological protection supervisor qualified after strict examination.
Surveillance of construction and performance of pre-operational tests.
Definite storage for the wastes.
Environmental monitoring.
Thoroughly controlled.
Only duly certified professionals.
Strict control throughout the stages required for the startup of a plant.
The workers involved are highly trained.
MEDIA
Surveillance in this area continues to be poor.
Inevitable damage to the population and the environment.
Lack of surveillance and control.
Scarce human resources and materials.
Precarious facilities.
Risks for the environment and public health.
The country does not exert due control.
Facilities without safety conditions.
Equipment is not inspected within terms.
No adequate equipment is available for surveillance.
Radioactivity is something uncontrollable.
pre-assumption


Why should nuclear technology be feared?
PRO-NUCLEAR
Man has always been exposed to natural radiation.
Nuclear techniques for environmental protection.
Reactors as the one in the Chernobyl plant were only exported to countries within the soviet block.
Monitoring evaluation, control and environmental recovery.
Irradiation does not contaminate.
Environmental preservation.
Any emissions in case of an accident are prevented.
MEDIA
The risk of accidents is a ghost that keeps hanging around.
Results can be dreadful.
Sensation of danger.
The country was surprised and scared after the disastrous case.
The danger of radioactivity.
Its effects can cause serious damage to plants and animals.
The most critical one, in Chernobyl, caused more than thirty casualties.
The number of contaminated people is inestimable.
Nuclear energy has to struggle with the uncomfortable problem or radioactive wastes.
Serious threat.
The world feels unsafe.
Radioactivity is destructive.
pre-assumption

Ducrot [2] asserts that “in order to understand the pre-assumption phenomenon, we need to relate it to the idea that the discourse [...] has a structure and that the preservation of the pre-assumptions is one of the laws defining such structure”. In the analyzed discourses, it has been found that certain pre-assumptions that are practically maxims with regard to the nuclear issue are maintained. Also, its negative action can be observed in the statements favoring the sector, weakening its structure and questioning the safety and the seriousness of its activities.

On the other hand, there is an anti-nuclear positioning in which use is made of the same pre-assumptions in order to build up a message that, usually, based on the catastrophes of the past, produces a stronger impact upon the public. This practice ends up strengthening the statement to which it opposes. Thus, if a discourse is not sufficiently strong to enforce its arguments, it is not in a condition to face the opposition successfully. If it is unsuccessful, it is because what it supports in its speech is useless as an instrument for attaining the desired objectives. Ducrot asserts that

> *[...] the refusal leads to a rejection of the dialog offered by the interlocutor while he/she speaks. Therefore, it also leads to not only accusing the adversary on false statements, but also of having behaved in an absurd manner. In fact, his/her speech, as any speech, implies the structure of a further dialog, and, for this reason, requires a sequence, opens up an exchange. However, at the same time, it imposes unacceptable conditions upon such exchange, rendering impossible the requested sequence, which was needed for its self-realization as a significant speech.*

This suggests that the pre-assumptions of the pro-nuclear discourse carry along the application of an unacceptable or doubtful speech for its target audience and, instead of obtaining a favorable reaction to what has been said, leading to a greater adherence of the public toward nuclear technology or to a better clarification on the issue, the sequence of the discourse is interrupted exactly by the fact that it contains elements that are considered as false and/or contradictory.

## 2. Perceptions and concessions

> *The course of the current history, between 1945 and nowadays, cannot be understood if the atomic issue is not taken into account along with its necessary supplement: missiles. Contrarily, the evolution of international events during the last 40 years can be traced by considering only the atomic factor, because the latter serves to explain the most important deeds within such period: the cold war, the thaw, the ease of tension, the attempts for cooperation between the East and the West, decolonization, the bipolarization of the current international society.* (GAJA *apud* MAIOCCHI [3])

The power of this scientific discovery gave way to an era based on the development of the nuclear power potential and on fear of both disasters and a world-scale holocaust —this having brought along political and economic consequences that were reflected in every continent. One of them was facing the reactions and the interests of the powerful oil industry, which was by no means willing to lose its market positioning, and the other, which is becoming more and more important, refers to the environment. In the latter, we come up with accidents in the plants and with the destination of wastes. No matter how scientists state that the risks are remote, the society has a different opinion. Actually, the population does not have many reasons for supporting the development of nuclear technology for peaceful purposes because its perception concerning the risks related to this sector is highly influenced by the negative historical context to which it inevitably leads.

Some relevant changes took place during the last few years and, today, the population is experiencing an *impasse* that ends up leading it to new considerations on this issue, even if does not have great affinity for it. Between a gradual waning of nuclear power plants, as it occurred in Germany and Sweden, and the expansion in the production of nuclear power in Finland. Between these two extremes, there are several societies —such as the Brazilian and the Argentine ones— experiencing a reactivation of their nuclear programs for power generation, and the Spanish one, whose efforts in finding solutions for the waste issue is quite well-known.

In addition, issues such as global warming, the increase in energy demand, the threat of an exhaustion of the traditional power generation potential and the movements against the construction of new hydroelectric plants because of their environmental impact are having an influence on public opinion and conditioning its position concerning nuclear energy, far more because of a need than because of its perception regarding risk. This means that, even if a given society does not view the production of nuclear energy favorably, it is willing to assume the risks as soon as the chaotic perspective of the greenhouse effect and the lack of electric power becomes evident.

In a research work performed by the Spanish *Empresa Nacional de Residuos Radiactivos* in 2001, the conclusion was reached that public opinion seems to be tinged by a position of the "no-to-nuclear,-but..." type referred to conditions such as need, safety and technical-scientific development. This is where the following concepts derive from: "nuclear energy is dangerous, but we cannot do without it"; "it is not clean, but it produces less atmospheric contamination than the other sources of energy"; "it is expensive, but it makes us less oil-dependent"; "there is never enough control over nuclear energy; however, its risks are controllable".

Also, in southeast Brazil, where the country's nuclear power plants are located, the same type of derived positioning occurred when, in 2001, rainfall dropped considerably and the prospects of a general blackout became possible. Society started to re-dimension its opinion with regard to the use of nuclear energy and began considering it as acceptable considering the serious problem to be faced by the country without the electricity provided by the hydroelectric plants.

This seems to lead to the conclusion that concessions, and even a conceptual change with regard to the nuclear issue, can arise when social perception faces different realities. At a time in which the whole world is facing a climatic change and important institutions as the United Nations alert about an actual hazard resulting from pollution, societies are starting to visualize the nuclear option as one of the most viable nowadays, even in the presence of the hot debate that this has generated.

Thus, the stigma experienced by this sector starts to evidence its "cracks". Through them, new opportunities may arise to face it with more appropriate arguments – aimed at the future, and not at the past.

## 3. What about the field of safeguards?

Would the institutions performing nuclear safeguards play a role in the discourse? Which is the dimension of such role, considering that these are the institutions offering some of the most important guarantees on what is said about the nuclear sector?

Up to now, we have mentioned the discursive activities of the institutions regulating and developing this technology, but would it be sensible to include the activities of the agencies in charge of safeguards? Would they be in a position that would also allow to approach society in this respect? Is there any work being carried out by these agencies that does not only look forward a transformation in the perception of risk by the lay public but also offers important instruments leading to a better understanding of the nuclear option?

The universe of this type of safeguards is quite critical and, consequently, there is hesitation concerning its actual possibilities of participation in the conflictive environment of the nuclear discourse. If, on one hand, these agencies represent exactly the live proof of the fact that nuclear power plants are under control and surveillance, providing a highly important guarantee to the population, on the other, such institutions possess technological information lending them a positioning of diplomatic and reserved character.

An interesting issue is considering that the safeguards work perspective and its systematic diffusion might help in clarifying public opinion or, at least, in eliminating the merely negative concepts regarding the nuclear issue. If the society is willing to accept risks when faced with a great need, perhaps, the way to face such risks could be based on other concepts providing new ways of visualizing this technology.

Considering that public opinion is, in fact, the one orienting important decision-making processes in society, it should not only be taken into account toward considering the nuclear option as a valid one, but, primarily, in order to select conscious and well-grounded options, no matter which. In this context, the role of safeguards appears as quite positive, leading to a mature and realistic path.

Since one of the sensitive points in the nuclear issue is precisely safety, it should be considered that those in charge of developing nuclear safeguards have the power to adopt a position before the public on the basis of the authority they were granted by representatives of society itself. In this direction, distrust concerning the possibilities for developing warlike material by the countries possessing this technology could be eased by the institutions in charge of safeguards in a closer relationship with the population. Of course, the risk of accidents or of deviation of nuclear material for non-peaceful purposes will always exist; however, it could be fit better, without groundless exaggerations or displays.

The discursive action by nuclear safeguards deserves reflection and should be taken into account considering that this is a critical moment for the whole world. In view of the seriousness of the problems being experienced and of the intensive debate that they are causing, this may be an excellent opportunity for demystifying public opinion and providing it with a landscape of new concepts on this issue.

## 4. Some starting points

It is important to bear in mind that we are not only working in a negative historical context. We are dealing with emotions and values. This is an abstract issue and it needs to be analyzed also at this level. Stating that the historical context plays a leading role in the permanence of the nuclear technology stigma is correct. However, when individuals are considered as agents who build reality in the world where they live, questions arise on the limits of historical actions and on the features of such individuals because, if, in fact, they are active, why is it that, often, they are unable to weigh nuclear technology in a more objective manner? What is this mechanism that exerts a certain “control” over significance, even knowing that there are multiple possibilities for signifying? Let’s go back to emotions, values and whatever leads human minds to produce sense.

An analysis of the cognitive bases (evidencing how 'cognitive spaces’ [5] intersect or reject each other) and the discursive bases (observing interdiscursivity [6] and its effects from sense [7]) regarding the issue under discussion will reveal the mechanism building the network of negative significances related to nuclear technology and allow to search for alternatives for such mechanism to be used in a different manner.

“Typically, we think about concepts as if sets of senses. We label them: *marriage, birth, death, strength, electricity, time, tomorrow*. Senses seem to be localized and stable. [...] However, [...] sense is not stored in a warehouse of concepts. By the contrary, it is alive and active, dynamic and distributed, built for local purposes of knowledge and action. Significances are not mental objects restricted to conceptual regions but, rather, complex operations of projection, linkage, connection, mixture and integration of multiple conceptual spaces.” Turner *apud* SALOMÃO [8].

Thus, it seems sensible stating that the polemics involving nuclear technology is not something so solid as it may appear. As whatever comes from human beings, it belongs to this movement that takes part of significance and that, by itself, exists in a thorough fluency. On the other hand, when the nuclear discourse is categorized within the notions of domain and projections of the cognitive theories, one can verify the extent in which social memory influences the processing of information about this technology, bringing memories of the tragedies occurred in this sector. When thinking is given in the field of significance related to any topic, previous knowledge is searched so as to apply the reasoning corresponding to the subject under analysis. It is this very type of reasoning the one structuring the unfavorable concepts about nuclear technology, because it implies actions by the social-historical context, the memory and sense processing, thus shaping —among other things— the images that lead to the stigma.

Fauconnier [9] states that human beings access and process the same information differently when they are in different contexts, adding that, if human cognition is so contextually configured, exploring

the types of connections that our minds tend to make and the types of effects produced by the various contexts is a crucial task. If we compare public opinion about nuclear technology ten years ago with what people think today, we will see that some variations have occurred deriving from problems that are leading the world population to a review of diverse concepts apparently crystallized and confirmed by the discourse of the media in general.

The importance of all the issues of the discourse discussed above becomes concrete in the extent in which we deal with a problem that occurs within a dynamic relationship between people and their institutions, as well as in individuals themselves in their interpretation. Living in a society means participating in a constant movement that leads society to exist, grow, evolve, signify and find its own significance, among other things. We must bear in mind that distrust regarding the nuclear issue occurs amid social practices, is submitted to them and exerts an influence upon them.

An approximation between this technology and society, based on a structured approach aware of the social and cognitive processes involved, is potentially liable to enlighten this issue with better grounded ideas. The past continues to be in fashion among the discursive practices concerning the nuclear issue and, perhaps, this will never change. Meanwhile, new actors have come on the stage and the way of dealing with this issue must be brought into the present.

If we acknowledge the power of public opinion in the decisions made concerning the nuclear issue, it appears as obvious that the discussions being held during the last few years with regard to energy exhaustion, global warming and environmental disasters, among other topics, will be conclusive in the acceptance —or rejection— of nuclear energy as one of the options in facing these problems. Undoubtedly, this is a delicate period and, at the same time, a gateway opened for the nuclear sector to produce a conceptual overturn about itself. For this reason, it is the time to go beyond what it seems to be and beyond what has been repeated by the institutions dealing with nuclear technology. It is also the time to think about the discursive potential of nuclear safeguards and on their capacity to play a decisive role in the construction, along with society, of new significances concerning this issue.

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6. Interdiscursivity. Notion in which, according to Dominique Maingueneau, a discourse “is born from work on other discourses”.

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# Networking for Safeguards Education

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***Abstract***

*For nuclear technology, which involves many disciplines of science and engineering, knowledge is one of the most important resources and needs to be managed carefully. Knowledge management consists of generating, disseminating, preserving and applying expert information from different sources in an organised way. The International Nuclear Safety Advisory Group to the IAEA Director General emphasized in Note No.4 of 2001 the importance of maintaining capabilities for nuclear research and education [1]. In the European Union, nuclear engineering education at twenty-two European universities was reinforced in 2003 with the European Nuclear Education Network (ENEN) association. The academic curriculum for this Master in Nuclear Engineering does not typically include the safeguards and non-proliferation aspects of nuclear systems (fuel cycle and reactors). A newly trained nuclear engineer, although during the work possibly exposed to relevant international agreements and regulations in the area of nuclear safeguards and non-proliferation (incl. the import/export of nuclear material and/or dual use goods…) thus generally has very little knowledge of the relevant treaties, their implementation, and their control.*
*Under the umbrella of ESARDA, a first training session on nuclear safeguards and non-proliferation was developed in 2004, which in the last 3 years has been further elaborated, resulting in a full week programme held in March 2007. The very valuable contribution and "ownership" of the various ESARDA working groups to the training course modules makes it a unique European initiative, led by the ESARDA Training and Knowledge Management Working Group. This course offers more than purely academic information, because it includes the various expertises of different actors in the nuclear field and is accompanied by exercises and laboratory visits. Not only the lecturers are representing the different European operators, inspectors, researchers, but also all ESARDA Working Groups are involved in the review of the lecture notes. The latter is used to establish a course syllabus as reference material for the full academic recognition of this course.*
*To deal with international nuclear security concerns many relevant initiatives have been launched in recent years. Also in the US and Russia, specific training and education programmes in nuclear safeguards and non-proliferation are built up and/or running including elements of nuclear security. The referred ESARDA training course also covers the links to these initiatives, through collaboration with the International Atomic Energy Agency and US labs and will be extended further to include Russian contributions.*

**Key-words**: Safeguards Education, Training, Knowledge Management

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## 1. Historical background

A declining knowledge and expertise in the nuclear field were reported e.g. by the OECD (2000) [2]. In 2002, a report of the Secretary-General of the United Nations [3] underlined that "there has never been a greater need for education in the areas of disarmament and non-proliferation" and that "Additional disarmament and non-proliferation related educational efforts are needed at all academic levels, for which support by the UN and its Member States is crucial". This report suggested 34 recommendations, of which implementation is only gradually proceeding.

In Europe a internationally recognized Master[2] of Science in Nuclear Engineering is offered at 22 European universities under the auspices of the European Nuclear Education Network Association. This education of nuclear engineers is appreciated mainly by the reactor industry, because it focuses on the labour market in the reactor design and operation. In this programme the subject of nuclear safeguards and non-proliferation is not addressed as a compulsory academic course. Few universities, such as Uppsala, Hamburg, and Florence are now starting to offer specialized courses and programmes in nuclear safeguards and non proliferation.

In Russia various Master degree programmes[3] dealing with the Fuel cycle and Nuclear Material Accountancy and Control are offered by the Moscow Engineering Physics Institute (Mephi State University) and the Tomsk Polytechnic University. " In the US nuclear material management and control is quite often embedded in the Faculties of Political Sciences and/or International Relations. To our knowledge a full fledged master degree which combines the scientific/technical with the political/legal aspects of nuclear safeguards and non proliferation is rare.

The European Safeguards Research and Development Association (ESARDA) has therefore built-up a short integrated programme, as explained below in detail. It is also addressing the problem of knowledge retention in the nuclear safeguards field [4]. The nuclear safeguards has been evolving over a relatively long period, learning from historical events and experts of different disciplines. In particular it needs a multinational collaboration and therefore international approach.

## 2. European Networks promoting Nuclear Education

The Joint Research Centre is connected on the one hand to European Nuclear Education Network (ENEN) by some personnel teaching some ENEN courses, and on the other to European Safeguards Research and Development Association (ESARDA) by its chairmanship of some Working Groups and its secretary role to the ESARDA. In the following a description is given of ENEN and the ESARDA WG TKM, because they are of direct benefit for "networking for safeguards education".

### 2.1. Results of the ENEN Association

The European Commission replied to the dwindling teaching capacity in nuclear science and technology by financing the set-up of European Nuclear Higher Education in a sustainable manner. The temporary European Nuclear Engineering Network, established through the EC 5th

[2] It is common practice to follow this programme as a Master after Master.
[3] Examples of such masters at Mephi are "Systems of Physical Protection of Nuclear Materials and installations", and "Automation of Power Physical Installations".

Framework programme project ENEN, was given a permanent character by the foundation of the ENEN Association, pursuing a pedagogic and scientific aim. [5]

Nowadays it includes twenty-two universities and thirteen other partners from nuclear industry, regulators and research centers from eighteen different EU Member States and every year successful students are granted a Master degree in Nuclear Engineering. The activities of the ENEN Association are organized in five committees: the Teaching and Academic Affairs Committee, the Advanced Courses and Research Committee, the Training and Industrial Projects Committee, the Quality Assurance Committee, and the Knowledge Management Committee. The ENEN envisages to provide a common qualification in nuclear engineering, with a mutual recognition and with a facilitated mobility of teachers and students. [6]

Barely fours years after being founded, the ENEN has completed a variety of tasks and delivered appreciated products to the European Higher Education by harmonizing nuclear education. The students with a ENEN diploma are highly valorized in the reactor industry. Nowadays the ENEN Association intends to expand its activities with professional training programmes.

## 2.2. Results of the ESARDA Working Group on Training & Knowledge Management

Since 2004 ESARDA started to introduce a compact course on Nuclear Safeguards and Non-Proliferation, which was since then yearly organized in March as reported in [7, 8]. In the meantime the programme of the course under the umbrella of ESARDA, as given in 2007 from Monday 5 March till Friday the 9th addresses:

- “what is safeguarded” (definition of nuclear material subject to safeguards),
- “where is such nuclear material found” (nuclear fuel cycle),
- “with which legal protective means” (the international and regional treaties, institutions and organisations),
- “how to control the nuclear material inventory and to audit an accountancy” (the methodology of verification, statistics for accountancy & control),
- “practical implementation of control measures” (how inspections are performed, and which tools the inspector has),
- “what additional information offers” (importance of the collection of open source data, illustrated with some case studies, and with import/export data control)

in a 80% core part with standard set of lectures, given by representatives from regulatory bodies (IAEA, IRSN, DG-TREN), industry (AREVA, BNG), and research (Stockholm University, Hamburg University, JRC-ITU, and JRC-IPSC) (cfr. Table 1). The remaining part is completed with topical lectures addressed by invited lecturers, such as from PNNL and IAEA in 2007 addressing physical protection, illicit trafficking, the Iraq case study, exercises, including satellite imagery interpretation etc. With this structure of a stable core part and a variable set of invited lectures, the course should remain sustainable and up-to-date. A course syllabus, as reference material for the full academic recognition, is under development.

| Institution | # lect. | Institution | # lect. |
|---|---|---|---|
| IAEA | 4 | AREVA (FR) | 2 |
| DG-TREN | 1 | BNG (GB) | 1 |
| Stockholm Uni. (SE) | 1 | JRC-IPSC | 5 |
| Hamburg Uni. (DE) | 1 | JRC-ITU | 1 |
| IRSN (FR) | 1 | PNNL (US) | 1 |

Table 1: Lecturers of the 2007 ESARDA course

The ESARDA 2007 course was followed by sixty students with eighteen different nationalities (cfr. Table 2) and eight have included the course in their academic curriculum and have it evaluated for the current academic year. The ENEN has recognized this 2007 ESARDA course academically for 3 points in the European Credit Transfer System.

As academic evaluation of the course those eight students of five different universities wrote an essay, paper and poster. The evaluation of the essays varied from E (Adequate) to B (Excellent).

| Institution | # stud. | Institution | # stud. |
|---|---|---|---|
| IAEA (UN) | 1O | King's Col. (GB) | 6E |
| CTBTO (UN) | 1E | London Econ. (GB) | 1L |
| IRSN (FR) | 2E | STUK (FI) | 2E |
| Hamburg/Freiburg Uni. (DE) | 8E | SKI (SV) | 1L |
| Nuc. Saf. Auth (SK) | 1E | ENEA (IT) | 1E |
| Tzochev NRA (BU) | 1E | Poli Torino (IT) | 7E |
| Budapest Uni (BU) | 1O | TU Vienna (AU) | 2E |
| Thrakom. (GR) | 1L | Omsk/Tomsk Uni (RF) | 2L, 2E |
| Uni Ghent (BE) | 2L, 3E | Torrejon SC (EU) | 2O |
| IISS UK (GB) | 1L | JRC IPSC/ITU (EC) | 12E |

Table 2: 60 students of 18 nationalities at the 2007 ESARDA course with different background (L=Law, Political Sciences, Int. Relations and Economics and History, E=Engineering, Physics and Chemistry, O=Others or unknown)

Four best essays with posters were selected for display at the ESARDA Symposium (Aix-en-Provence, 22-24 May 2007):

1. Vienna University of Technology: Ultra-low-level measurements of argon, krypton and radioxenon for treaty verification purposes
2. Tomsk Polytechnic University: Control of nuclear material hold-up in process lines and equipment at isotope-separation facilities
3. Graz University of Technology: Methods for the detection of undeclared plutonium production facilities
4. University Ghent: Advantages and disadvantages of fusion-fission based hybrid reactors

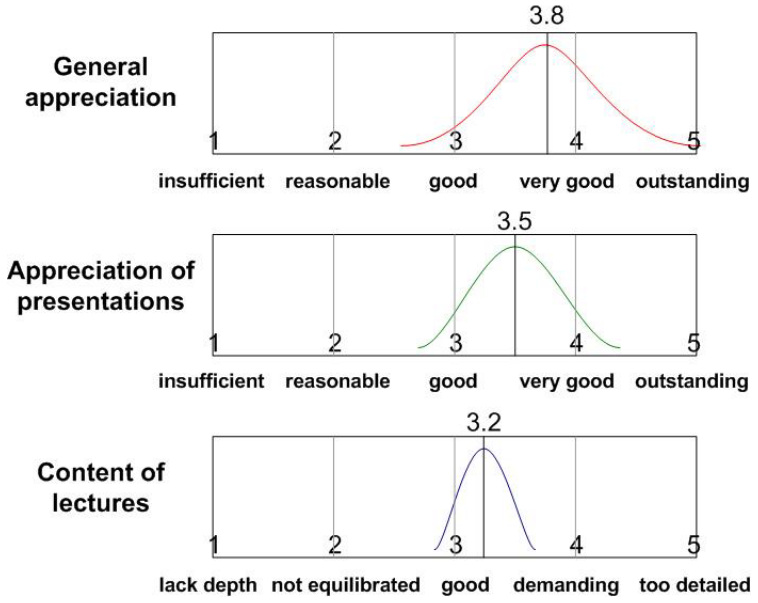


Fig. 1: Appreciation of the ESARDA Course (Ispra, March 2007)

The course feedback allowed to perform only a limited statistical analysis but the remarks of the students were very valuable to draw few lessons. In particular the specific examples or applications addressed in or after the lectures, and the practical visits to the 5 labs, (one per day) were appreciated very much.

The students could mark the general appreciation of the course from 1 “insufficient” to 5 “outstanding”. The outcome of the appreciations is presented in Fig. 1. A mean value of 3.8 was scored with standard deviation of 0.7. This means that the overall opinion is near 4 “very good”. This is a clear indication that the course is on the right track and responds appropriately to what students are looking for, nevertheless their varying background. Therefore the course will be repeated on a yearly basis in the same way.

Major comments and suggestions of the students indicated the wish for further practical applications, and more time dedicated to the class-room exercise. The exercise, presented by IAEA, asked the students to take the role of a safeguards inspector. The description of a fictitious state bordering a nuclear weapon state, “Salinas” was provided, that has a nuclear power plant, a research reactor, and a fuel fabrication plant, and that signed recently NPT and concluded a comprehensive safeguards agreement with the IAEA. A few more info’s were provided under the description of the case. The job of each group of students consisted of designing a safeguards approach, keeping in mind technical capabilities of the country, minimal disruption to the operator and the cost of inspections. The outcome was discussed between the different groups and the IAEA lecturer in a dedicated afternoon session. In the next course of March 2008, we envisage a two-phase approach, with a feedback to the students after a first rough design and a splitting up of the more specific tasks between the different student groups.

## 3. Integrated Knowledge Management as Perspective

### 3.1. Management of nuclear safeguards knowledge by and within ESARDA

The ESARDA Nuclear Safeguards and Non-Proliferation course is offering more than a standard academic course, because the course contains contributions of different experts in the field. The lecturers are selected by the ESARDA WG TKM such that there is a representation from the different ESARDA members: including European nuclear industry, EURATOM and IAEA inspectorate, European researchers.

The syllabus is under development with the lecture note contributions form the authors and the review and completion of all ESARDA Working Groups. The label “ESARDA approved” to the syllabus ensures that it covers an overview verified by the recognized experts and presented under a European view shared by the different ESARDA actors. The syllabus will be published as a Special ESARDA Bulletin. With the course schedule, the syllabus and evaluation scheme a standard recognition at ENEN level can be asked on a long-term basis.

The ESARDA WG TKM is accomplishing with the course and syllabus a first task of information gathering and dissemination. This knowledge gathering and structuring needs to be continued as support to a future integrated knowledge management. The latter underlines the second part of the WG’s name “Training and *Knowledge Management*” with the role of knowledge retention, including collection, structuring, dissemination and assessing the application.

### 3.2. Extension towards training and nuclear security

In recent years, nuclear and other radioactive material and associated facilities and transports have been identified as requiring higher levels of control, and nuclear threat scenarios have become more dangerous and more complex. At the same time, energy security concerns and fears of climate change are making nuclear power more attractive. In this context, States have to be prepared to acquire the necessary knowledge and skills to fulfill adequately their obligations stemming from adherence to the international instruments related to nuclear security. Several binding and non-binding international instruments such as the *Convention on the Physical Protection of Nuclear Material,* the *Convention on the Suppression of Acts of Nuclear Terrorism, UNSC Resolution 1540* or the *Code of Conduct on the Safety and Security of Radioactive*

*Sources,* require, inter alia, States to establish appropriate systems to prevent, detect and respond to unauthorized acts involving nuclear and other radioactive material. In order to meet these requirements, an increased attention to education and training in this field is crucial.

In the light of the identified need for better international cooperation and interaction in nuclear security education and training, the IAEA convened at the end of April 2007 a consultants meeting to explore the possibility of better coordination in this field. This was in follow-up of a recommendation of the International Seminar on Education & Training in Non-Proliferation, Security Culture & Accounting for and Control of Nuclear Material, organized by CNS in Moscow, in November 2006.

This international coordinating mechanism will address key issues such as coordination of nuclear security educational and training activities at international level in an effort to avoid overlaps, to ensure the effective use of resources and to identify gaps in the existing programs.

As a first step it was suggested conducting training needs analysis in order to identify the target audience and their required competency profile. This target audience ranges from the decision makers, politicians, diplomats, managers, CEO as well as technicians, border guards to the worker in the field, experts (lawyers, engineers, …) or in the control loop (regulatory body, safety authorities,…). They may be starting a career or be already in senior positions, they may work in the nuclear area as specialists, or through cross cutting activities.

The second step would be to consider training as a tool for career management. Training can be a career development tool as well as a component in knowledge management. Training could also assist technical staff to obtain a broader knowledge outside or their normal duties. Short training courses (e.g. two-day' sessions) could also become mandatory for senior management with nuclear responsibilities, such as nuclear crisis management.

The third step would be to assess the effectiveness of training courses in States. Setting up a relevant feedback mechanism, gathering information from the "field" (e.g. events related to illicit trafficking, communication difficulties between National Authorities and Operators, lessons learned from Safeguards related events). Analysis of this data will lead to better understanding of States' training needs related to nuclear security issues.

Through enhanced co-operation between different international organizations, better use will be made of dwindling teaching capacity, scientific equipment and research infrastructure to ensure the efficient utilization of limited resources.

## 4. Conclusions

The ESARDA course is a contribution in setting up through a network the necessary tools meeting the increasing education needs in the nuclear safeguards and non-proliferation area and beyond:

- Self sufficient, it provides operating engineers, national SG authorities representatives, young professionals or qualified managers with the necessary knowledge of the relevant international agreements and regulations in the area of nuclear safeguards and non-proliferation and of the tools to monitor those.
- Recognized as an optional academic course in the European Master Degree curriculum for nuclear engineering, it is fully included in the European educational system, as initial education or in the framework of continuing training.
- Self-sustainable, it is setup with a core part, lectured by a pool of lecturers representing ESARDA members and kept up-to-date with specific topics given by invitees, allowing collaborations with the IAEA, US national laboratories and Russian institutions. By

enabling a pan-European and even world-wide participation of students and young professionals an international view on today's non-proliferation regime is given.
- Gathering lecturers from concerned international, national or private organisations, it represents a real focal point where experiences can be shared
- Fed by the ESARDA Training and Knowledge Management Working Group in collaboration with the other Working Groups, its up-dating is granted, and it makes possible capturing the knowledge in Safeguards from various horizons, including building in a return of experience
- Its information gathering, structuring and validation process can constitute a contribution towards future more comprehensive knowledge management initiatives.

## 5. Acknowledgement

The authors acknowledge the members of the ESARDA WG TKM and all lecturers and laboratory responsibles for their efforts, enthousiasm, support to establish a sustainable ESARDA course. The contributions of the other ESARDA Working Groups are also acknowledged, as is the support of the ESARDA Secretary.

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# Development of a Quality Management System (QMS) Training Package

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***Abstract:***

*To support the implementation of a department-wide Quality Management System (QMS), the IAEA Section for Standardization, Division of Concepts and Planning, Department of Safeguards commissioned the Canadian Safeguards Support Program (CSSP) to develop an interactive computer-based training course (ICBT) on Quality Management to introduce new and existing staff to basic QMS principles. Since January, 2006, a CSSP project manager, instructional designer, and programmer/graphic designer have been working with subject matter experts in the Department of Safeguards Division of Concepts and Planning to research and develop the course. The course, which is now complete, is scheduled for delivery throughout the department beginning in mid-2007. The course itself, and the step-by-step process used to develop it, are intended to serve as models for a curriculum of QMS and other courses to be delivered to IAEA staff at their desktops by means of the Agency's LAN.*

**Keywords:** quality management; QMS; ISO 9001: 2000; e-learning; computer-based training

## 1. Introduction

The IAEA Department of Safeguards is implementing a Quality Management System (QMS) as a means of ensuring soundly-based safeguards conclusions. Guided by a Quality Policy Statement approved by senior management in November, 2004, the Departmental Quality Manager in the Division of Concepts and Planning, Section for Standardization, has since begun to implement a wide range of measures designed to create a QMS based on the requirements of the ISO 9001: 2000 standard.

An early priority has been the development of a training program to introduce staff to the QMS and to teach specific groups how to apply quality management principles to their jobs. The Department of Safeguards has been providing Quality Management training to the staff, with the current curriculum consisting of a one-hour introductory seminar for new inspectors, a 1.5 hour briefing for new staff, a two-day seminar for safeguards managers, and a five-day workshop teaching tools and methodology for staff intending to work on continual process improvement of Agency work processes. In addition, it was determined that an interactive computer-based training course (ICBT) on Quality Management is required to introduce all new and existing staff to basic QMS principles. Offering computer-based training at all desktops is desirable to introduce new staff to QMS concepts as early in their employment as possible and to provide easy refresher training for any staff members who wish to take the course at their desk at their convenience. ICBT complements classroom training as people have different learning styles and preferences and this format may be easier for some to learn from than from conventional classroom training. In addition, ICBT can be used to provide updated information to staff on developments in a more timely manner.

To carry out this work, the Canadian Safeguards Support Program (CSSP) was approached for assistance. Previously, the CSSP had developed several successful computer-based training programs for the Department of Safeguards on topics such as *CANDU Power Station Fundamentals*, *VXI Integrated Fuel Monitor (VIFM) Procedures*, *Interpreting Bundle Counter Radiation Signatures,* and *Core Discharge Monitor Radiation Signatures*. In January, 2006, a CSSP project team consisting of a project manager, instructional designer, and programmer/graphic designer was created to work with subject matter experts in the Department of Safeguards Division of Concepts and Planning.

## 2. Objectives and Requirements

The main objective for a computer-based Quality Management training course was to explain to new staff how the QMS works and what impacts it will have on how they carry out their duties. Since communication is a key element of a QMS and it is essential that staff be aware of the development, implementation and functioning of the Departmental QMS and their role in the functioning of the QMS.

More specifically, the Department's objectives for the course are that after completing the course participants should be able to:

- Describe what is meant by quality and a quality management system;
- Identify the key stakeholders who are interested in the quality of the Department's work;
- Describe the implementation of the Department's QMS;
- Describe how the QMS impacts on their work; and
- Know where to find further assistance and information.

In addition, the required characteristics of the interactive training program should:

- Require no more than 2 hours to complete;
- Permit the on-line assessment of the participant's understanding of the learning materials;
- Incorporate a variety of media and avoid a "page-turner" style of presentation;
- Allow some components of the program to be easily updated;
- Be accessible from IAEA desktop computers via the Agency's local area network (LAN);

- Lead participants through the learning materials systematically, ensuring that they master one set of concepts before moving on to the next set; and
- Generate a record of staff members who have taken and passed the course, and make this information available in an organized format to authorized personnel.

The production team undertook the task to pursue the development of a highly interactive, media-rich, centrally administered training program delivered over the IAEA Intranet, as a convenient form of presentation to staff.

## 3. Content Capture

As the Department of Safeguards was already in its second year of QMS implementation at the time the ICBT was commissioned, a number of information resources were readily available to the development team. These included a comprehensive QMS Web site developed by the Department as a reference resource for staff, instructional materials developed for classroom-based courses, a number of PowerPoint presentations, conference papers, and a library of books and articles on subjects related to quality management.

The developers were also given access to Department of Safeguards subject matter experts (SMEs), who made themselves available for extended face-to-face and telephonic interviews. The interviews proved to be particularly helpful in understanding the operation of the QMS within the Department's unique operating environment.

An important feature of the training modules is the video commentary by a senior staff member that introduces each module. While each commentary was scripted by the Ottawa-based production team in consultation with the individuals who would appear on screen, arrangements were made with the Department of Safeguards training department to use their video equipment and crews to capture the commentaries. This collaborative effort is believed to be efficient and cost effective.

## 4. Instructional Design

The resulting computer-based training package, entitled *The Road to Safeguards Quality: An E-Learning Tutorial*, includes the following presentation techniques:

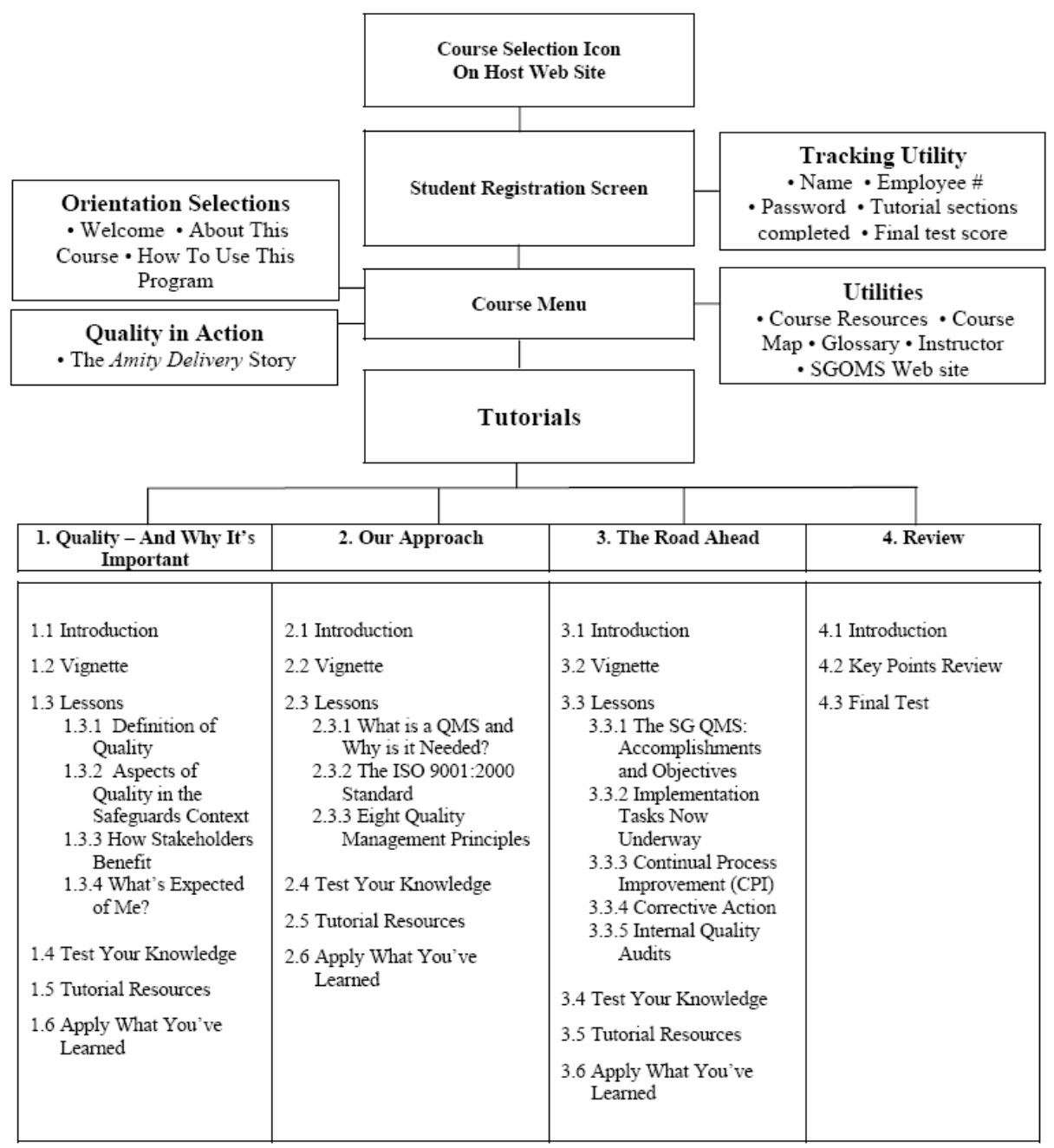
- Short video clips featuring testimonials and scene-setting commentaries by experienced QMS practitioners throughout the program;
- A "story" module designed to provide a light introduction to key QMS concepts;
- Animated vignettes at the beginning of each module to capture key messages;
- Frequent quizzes and exercises to test understanding and broaden perspectives;
- Continuous online access to reference documents and Web resources;
- Ready access to departmental trainers and subject matter experts by e-mail;
- Tracking of progress to help participants resume their studies from session to session;
- A modular technical design that facilitates course updates; and
- A pass-fail test to conclude the course.

Based on e-learning best practices, the tutorials are characterized by:

- Short tutorial duration (15-20 minutes would be needed for a typical participant to cover all of the materials in a tutorial);
- Lessons within tutorials limited to a maximum of 12-15 screens, each featuring a main teaching point;
- Learning objectives introduced at the start of each tutorial;
- Key points for each tutorial highlighted in a introductory vignette;
- Graphic illustration of teaching points wherever possible;
- Frequent use of examples and demonstrations;
- Frequent requirement for learners to interact with the training package (e.g. by answering a question, launching a demonstration, assembling a diagram, etc.); and

- Immediate feedback to questions, quizzes and exercises.

The following diagram captures the organization of the ICBT as it finally emerged.




## 5. Interface Design

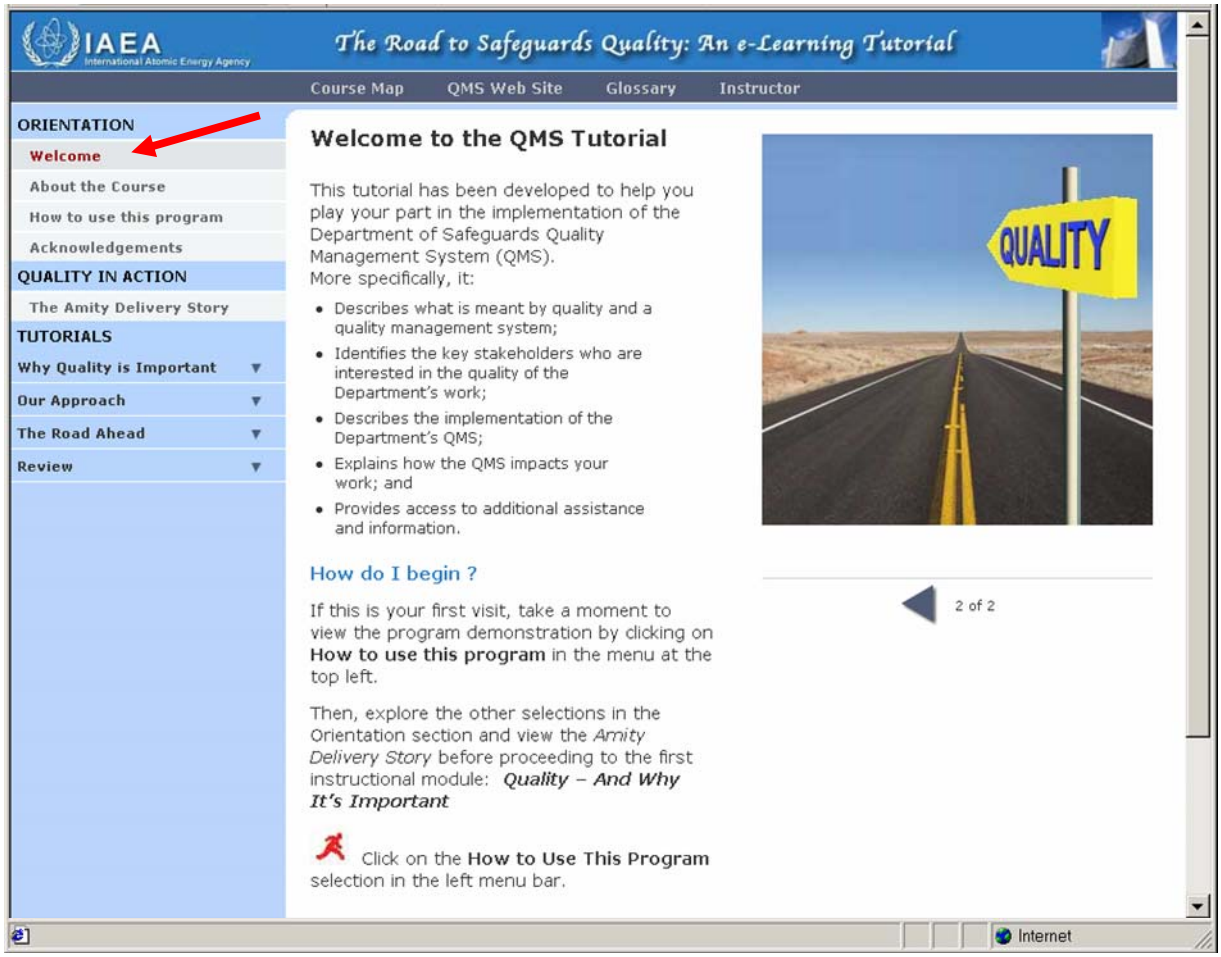
The IAEA's Visual Identity Specifications (such as fonts for printed materials, a color palette, and the use and treatment of the IAEA logo) were followed for the development of this program. Working within the specifications, the graphic designer began by developing a look-and-feel for several types of screens, two of which are illustrated below.

## 5.1. The Course Menu

The Course Menu consists of:

1. **IAEA Identifier and Course Title Banner**
2. **Utilities** (Course Resources, My Progress, Glossary, Instructor, QMS Home Page)
3. **Orientation Selections** (Welcome, About This Course, How To Use This Program)
4. **Numbered Tutorials** (major topics within the course, with drop-down menus for sub-selections)
5. **Main Presentation Window** (where text and illustrations associated with the current selection are displayed)

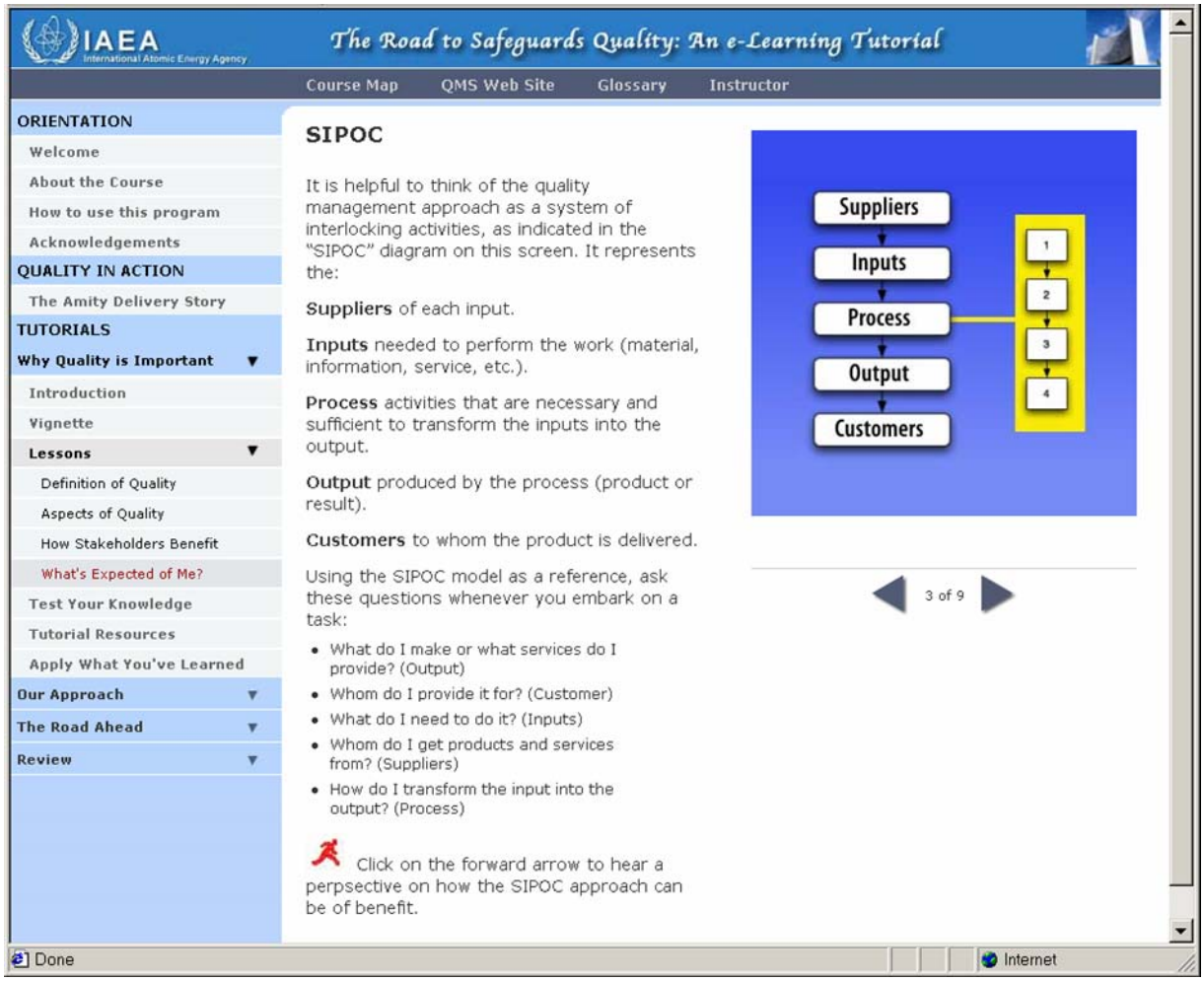
The *Welcome* selection appears on screen as the default. The current selection is always highlighted on the menu.




## 5.2. The Presentation Screen

Lessons are divided into several numbered screens, each covering one teaching point. A presentation screen will normally consist of a text block, underlined hyperlinks (for definitions, links to reference documents, etc.), and an illustrative "visual" (e.g. a photograph, chart, diagram, animated sequence, etc.)

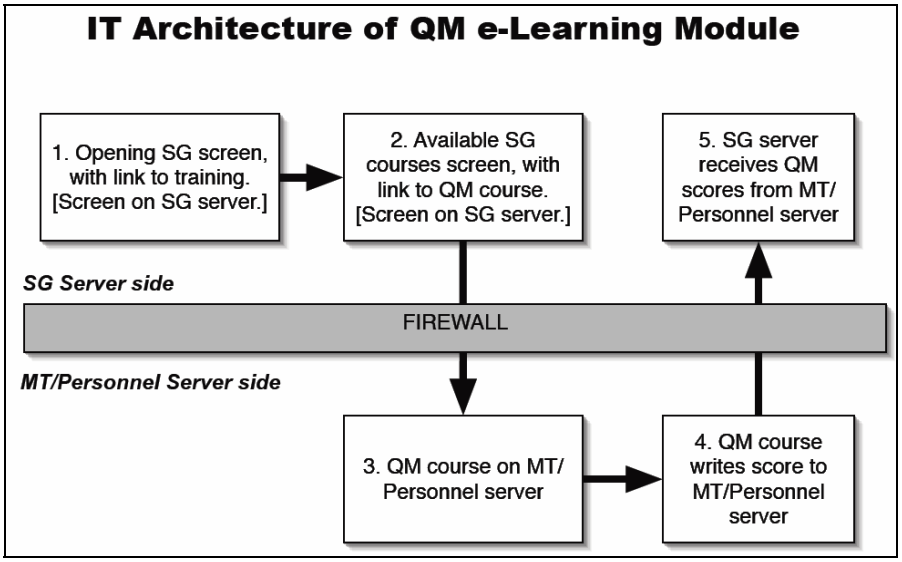
The elements of a *Presentation Screen* – this one illustrating the SIPOC concept (**S**uppliers – **I**nputs – **P**rocess – **O**utput – **C**ustomers) are shown below.




## 6. Technical Requirements & Issues

The decision to deliver the course by means of the IAEA Intranet presented several challenges arising from: a departmental Intranet architecture that, for security reasons, includes firewalls that limit outside access to Department of Safeguards materials; and the need to first implement a learning management system for online courses.

The contractor is working with Departmental IT staff to identify ways to overcome firewall barriers, without compromising security, to make the program available outside the IAEA Department of Safeguards. The learning management system implemented for this course would support a growing curriculum of on-line courses in coming years.




In addition, to meet the specifications required for the training package, there was a need to implement technical features that would permit the tracking of key user data (e.g. names/passwords of registrants, cumulative figures on course/tutorial usage), as well as the tracking of individual performance such as each user's progress through the course, and final test score to confirm that the minimum standard has been achieved. For privacy protection, this kind of information is made available only to authorized personnel.

## 7. Effectiveness Verification

End-user evaluations were considered to be important throughout the design, development and implementation cycle to ensure that the finished product is perceived to be attractive, intuitive, functional and useful from the user's point of view. Having course documents, interface designs and prototypes available online provided a means of continuous evaluation and feedback by the Department's team. On two occasions during the development phase, the Agency's QMS team arranged for several typical end users to interact with test versions of the program and to provide comments.

A particularly important validation took place in early February, 2007, when a group of typical users assembled in an IAEA training room equipped with multimedia-and-Internet-capable computer workstations like the ones in their offices. There were eight participants in the focus group session, representing a broad cross-section of experiences in the Department.

Following a brief introduction, the participants were asked to go through the course materials from start to finish. Approximately 2 hours were allowed for this task. Actual completion times ranged from about 1 hour for a participant who had worked in the QMS area for several years, to more than two hours for a participant who had only recently joined the department. A typical completion time, which included the final test, was approximately 2 hours. The entire group then participated in a discussion session in which comments were invited on the course's appearance, functionality and learning materials. Each participant also filled out a one-page questionnaire which permitted him or her to rank certain aspects of the program and to provide written comments. These were later incorporated into final changes to the program before it was released for general use.

Significantly, the average test score was about 89% – well above the 80% standard set by the Department.

## 8. Conclusion

In many ways, the Road to Safeguards Training e-learning course is a prototype for other on-line training to follow. So, it is important to understand what went right and what went wrong, and to capture best practices. After reviewing the experience, we attribute the very positive outcomes to:

- The clear objectives for the course set at the outset by the client;
- Time taken at the beginning to understand the information needs of employees, and to understand the constraints that would affect their ability to complete an on-line learning program successfully from their desktop computers;
- The extensive, well-organized reference materials provided to the production team by the client;
- Continuous access by the production team to QMS subject matter experts to clarify points of information and to get reaction to presentation approaches;
- Ongoing assistance from Agency IT staff from the earliest days of course development to define technical constraints and to overcome potential delivery problems; and
- Periodic review of course materials by end-users to ensure that the program was achieving its intended objectives.

After an initial period of heavy usage during which all departmental staff will complete the QMS training module, it is estimated that up to 40 new staff members recruited into the Department each year will complete the *Road to Safeguards Quality* course during their first month of employment. The lessons learned from this seminal course development experience should be useful to other on-line training programs in the years ahead.

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# A SERIES OF COMPREHENSIVE U.S.-CHINA NDA WORKSHOPS TO STRENGTHEN SAFEGUARDS CAPABILITY IN CHINA

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## *Abstract:*

*The U.S. Department of Energy's (DOE's) National Nuclear Security Administration (NNSA) and the China Atomic Energy Authority (CAEA) collaborated to support an Integrated Nuclear Safeguards and Security (INSS) workshop between the U.S. and China. The INSS workshop promoted the development of a regulatory infrastructure for, and the implementation of, best practices for nuclear material protection, control, and accountability as well as the development of international safeguards to strengthen the safeguards capability of China. As part of the INSS workshop, NNSA and CAEA conducted a series of comprehensive nondestructive assay (NDA) workshops in China and the U.S. from October 9 through November 3, 2006. More than 36 Chinese participants from various organizations and nuclear facilities attended the first of the NDA workshops, from October 9 to 13, at the CIAE in Beijing. This workshop included fundamental NDA measurement principles, techniques, and instruments, as well as their implementation in the nuclear fuel cycle, process holdup, wastes, and unattended remote monitoring systems, supporting both domestic and international safeguards. We performed laboratory demonstrations and exercises made up of uranium enrichment measurements, plutonium isotopic analysis, neutron coincidence counting, and holdup measurements. After the Beijing workshop, seven selected Chinese technical experts traveled to LANL from October 17 to 24 to extend their fundamental NDA knowledge and measurement experience. The LANL workshop included lectures and state-of-the-art laboratory exercises using gamma-ray, neutron, and calorimeter techniques. Participants then traveled to ORNL on October 25–27 for extensive, hands-on training in process holdup measurements. While in Oak Ridge, from October 29 to November 1, the participants attended the International Workshop on Best Practices for Material Holdup Monitoring, sponsored by the Institute of Nuclear Materials Management (INMM), to learn additional best practices for holdup measurements and to participate in joint discussions with experts and practitioners from various fuel-cycle facilities around the world. The workshops concluded with a technical tour of the gaseous diffusion enrichment plant in Paducah, Kentucky, on November 2 to observe how NDA measurements are conducted at an operating, civil nuclear facility in the U.S. to support domestic regulatory requirements for nuclear material control and accountability. This paper provides details on the modules, results, experiences, and lessons learned from the workshops.*

**Keywords:** Safeguards; NDA; Training; Workshop

## 1. Introduction

The United States Department of Energy/National Nuclear Security Administration (DOE/NNSA) and the China Atomic Energy Authority (CAEA) collaborated to conduct a series of nondestructive assay (NDA) workshops in China and the U.S. from October 9 through November 3, 2006. This effort was part of the U.S.-China Integrated Nuclear Safeguards and Security (INSS) Workshop. The INSS workshop promoted the development of a regulatory infrastructure for, and implementation of, best practices for nuclear material protection, control and accountability, and international safeguards under the *Agreement Between the Department of Energy of the United States of America and the State Development Planning Commission of the People's Republic of China on Cooperation Concerning Peaceful Uses of Nuclear Technology* (PUNT).

We conducted the first workshop from October 9–13, 2006, at the China Institute of Atomic Energy (CIAE) in Beijing, China. More than 36 Chinese participants from various organizations and facilities attended the first week of the workshop. The workshop contents included fundamental NDA measurement principles, techniques, and instruments, as well as their implementation of the nuclear fuel cycle, process holdup, wastes, and unattended remote monitoring systems in domestic and international safeguards. Laboratory demonstrations and exercises were performed on uranium enrichment measurement, plutonium isotopic analysis, neutron coincidence counting, and holdup measurement.

Following the NDA workshop held in Beijing, seven selected Chinese technical experts traveled to the U.S. to extend their NDA knowledge and measurement experience at the Los Alamos National Laboratory (LANL) and the Oak Ridge National Laboratory (ORNL). The LANL Workshop from October 17-24 contained well-designed lectures and state-of-the-art laboratory exercises on gamma-ray, neutron, and calorimeter techniques. The participants traveled to ORNL for extensive hands-on training on process holdup measurements on October 25–27. The exercise used seven fixtures to simulate pipe and duct assemblies in a generic processing facility. They also attended the International Workshop on Best Practices for Material Holdup Monitoring, sponsored by the Institute of Nuclear Materials Management (INMM) held at ORNL from October 29 through November 1, 2006. The Chinese delegation was introduced to additional best practices for holdup measurements. They also participated in joint discussions with experts and practitioners from various fuel cycle facilities around the world. The workshop concluded with a technical tour of the gaseous diffusion enrichment plant, in Paducah, Kentucky, on November 2 to observe how NDA measurements are conducted to support domestic regulatory requirements for nuclear material control and accountability at an operating civil nuclear facility in the United States.

This paper reports details on workshop activities, results, experiences, and lessons learned from the workshops.

## 2. Beijing Workshop At CIAE

We conducted the first NDA workshop, as part of the INSS workshop, from October 9–13, 2006, at the CIAE in Beijing, China. Figure 1 shows all instructors and participants. The purpose of the Beijing workshop was in preparation for seven Chinese NDA specialists who would come to the U.S. for extensive NDA training under the INSS Workshop as an NDA Group and would provide the opportunity to other participants who will be unable to join the NDA Group for training in the U.S. More than 36 Chinese participants from various organizations and facilities including CIAE, the Chinese Academy of Engineering Physics (CAEP), China National Nuclear Corp (CNNC) Everclean, Beijing Institute of Nuclear Engineering, Lanzhou Enrichment Plant, Shaanxi Enrichment Plant, and the China North Nuclear Fuel Corporation attended the first week of the workshop from October 9–13, 2006, at the CIAE in Beijing, China. The workshop contents

included fundamental NDA measurement principles, techniques, and instruments, as well as their implementation with the nuclear fuel cycle, process holdup, waste measurements, and


Figure 1. NDA Workshop at CIAE, Beijing, China

unattended remote monitoring systems in domestic and international safeguards. Laboratory demonstrations and exercises on uranium enrichment measurement, plutonium isotopic analysis using high-resolution gamma-ray spectroscopy, neutron coincidence counting and software, and holdup measurement were performed. For process holdup measurements, we presented basics and measurement techniques with automation of holdup measurements, including hardware and software. To create an interest in the recently promoted Global Nuclear Energy Partnership (GNEP) program, a special lecture on the safeguardability of pyrochemical reprocessing was presented and discussed. Figure 2 shows some of the activities carried out during the CIAE Workshop.

Figure 2. Some activities during the CIAE Workshop

A panel discussion session was held to conclude the workshop. Topics were well received by the Chinese attendees, and interest was high. More importantly, the workshop focused attention on

NDA Group delegates of the INSS Workshop so that training could continue almost seamlessly when they visited LANL (fundamental NDA training) and ORNL (process holdup training) in the following three weeks.

## 3. Los Alamos Workshop at LANL

Following the NDA workshop held in Beijing, seven selected Chinese technical experts from CIAE, CAEP, the Shaanxi Enrichment Plant, and the Lanzhou Enrichment Plant traveled to the U.S. to extend their NDA knowledge and measurement experience at LANL and ORNL. The LANL Workshop (Figure 3) from October 17–24 contained well-designed lectures and state-of-the-art laboratory exercises on gamma-ray, neutron, and calorimeter techniques. In preparing the participants of the INSS Workshop for work in the field as safeguards practitioners, the workshop consisted of both lectures and hands-on laboratory training with significant amounts of special nuclear material. For the laboratory part of the training, the participants were divided into two or three groups to work individually with an instructor on an NDA instrument.

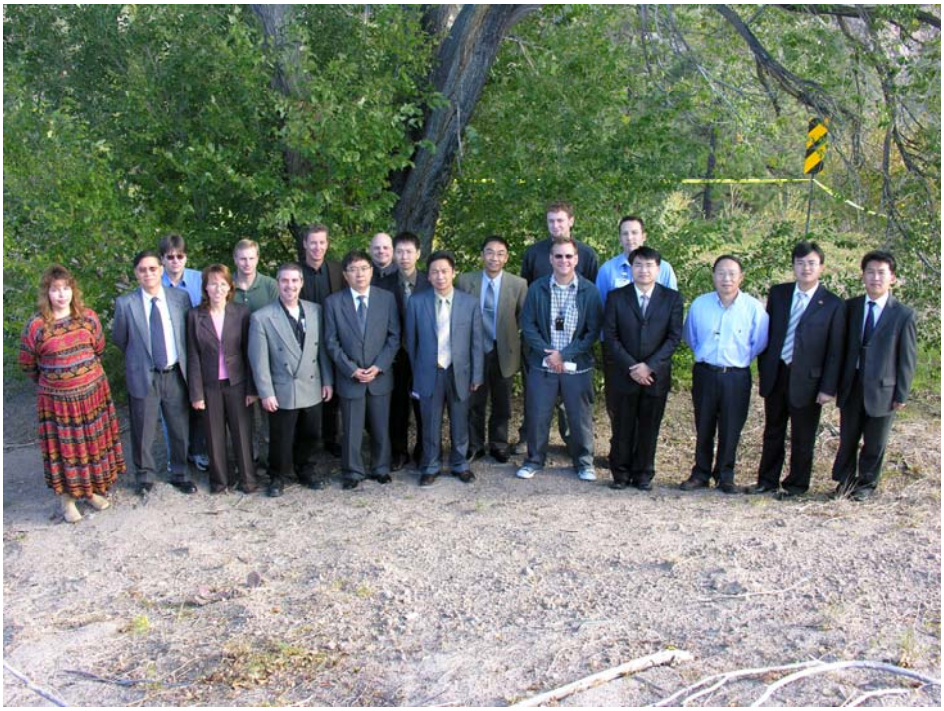

Figure 3. NDA Workshop at LANL

### 3.1. Gamma-Ray NDA

The content of the gamma-ray part of the workshop consisted of basic gamma laboratory, gamma ray spectroscopy, and gamma-ray isotopic analysis with a high purity germanium (HPGe) detector system and the Fixed Energy Response Function Analysis with Multiple Efficiencies (FRAM) isotopic analysis code.

In preparing the participants for work in the field as safeguards practitioners, a conveyance of the fundamental concepts of gamma ray spectroscopy must be accomplished. To this end, this lecture provided a cursory survey of gamma-ray spectroscopy that included but was not limited to the basic physics of the origin and nature of gamma rays, key safeguards isotopic signatures and branching ratios, the interaction of gamma rays with matter and the spectral features these

interactions produce, gamma ray detector systems and the effects of their limitations, and introductory statistical principles of measuring random nuclear processes. This lecture was designed to provide the background material for lectures and laboratory exercises covering fundamental gamma ray spectroscopy techniques as well as the use of the FRAM isotopic analysis code.

In addition to a review of general concepts, specific attention was given to the detector hardware and software that would be used in the exercises. The participants made use of an HPGe detector, the ORTEC DSPEC Plus MCA, and the ORTEC data acquisition software Maestro. The general criteria for optimal configuration of the HPGe detector system was covered as well as the particular steps of how this optimization was achieved as illustrated through Maestro screenshots.

The intent of the introductory laboratory exercise (Figure 4) was to provide an understanding of the fundamental concepts of gamma-ray spectroscopy through hands-on measurements with an HPGe detector system. During this exercise, the students became familiar with the operation and optimization of the multichannel analyzer and its associated components. An oscilloscope was made available to illustrate the time dependence of the signals from the preamplifier. The basic features of a monoenergetic gamma-ray spectrum were made evident with the use of a $^{137}$Cs source. Also, the dependence of counting statistics on system dead time was studied.

The lecture on the FRAM isotopic analysis provided an overview of the FRAM Plutonium Isotopic Analysis code. The topics included ranges of application with regard to energy, shielding, material, and detector type. Also covered is how FRAM works in general, including the internal calibration measures taken as well as how the analysis is performed. A cursory treatment of how to use FRAM was given to let the participants have some familiarity with the system before the FRAM laboratory. In the FRAM laboratory, the participants learned to use FRAM by assaying a variety of plutonium and uranium-bearing items. Exercises included basic data acquisition and analysis with FRAM, a study of resolution and throughput versus dead-time, applying proper FRAM parameter sets, depending on measurement conditions (such as the presence of different absorbers), and how to handle nonstandard energy calibrations.

Figure 4. Gamma laboratory at LANL

## 3.2. Neutron NDA

Two days were spent on neutron NDA-based techniques. The goal of the training was to review the basic materials in neutron-based NDA, introduce advanced techniques (multiplicity counting), and to provide a hands-on measurement experience using special nuclear material. The training covered the principles of how neutrons interact with matter, the designed features of neutron counters, electronics used for neutron counting, and analysis methods, including advantages and disadvantages.

Lectures and laboratory exercises of neutron NDA techniques included a review of basic neutron counting, active neutron coincidence counting, passive coincidence, and multiplicity counting

(Figure 5). Basic neutron counting discussed counting statistics, detector design and setup, neutron absorption, neutron scattering, and neutron shielding. The instrument used for the basic neutron counting laboratory was the 2-Tube Slab Detector.

Figure 5. Neutron lecture and laboratory at LANL

From the coincidence counting lecture and laboratory sessions, participants learned to use shift register coincidence electronics to perform measurements on several types of neutron sources, to see how the raw data are obtained, how coincidence rates and errors are calculated, how coincidence counts are affected by a random neutron background, and how they are affected by matrix materials. Participants also determined some of the important detector parameters that are needed for assay measurements, such as deadtime, efficiency, gate fraction, die-away time, and matrix materials. Finally, participants started to use the International Neutron Coincidence Counting (INCC) program for data collection. INCC is a Windows program designed for general-purpose passive and active neutron assay applications.

In the active neutron assay session, students used the Active Well Coincidence Counter (AWCC) to assay varying enriched oxide samples. Because the spontaneous fission rate of the uranium isotopes is very low, the AWCC uses americium-lithium (AmLi) neutron sources to induce fissions in the uranium. The coincidence neutron rate from the induced fissions is counted with the shift register electronics. Participants determined a calibration curve for uranium oxide in the fast mode and then perform assays using this calibration curve.

The instructional goals for the passive coincidence and multiplicity counting assay session is to understand the basics of passive neutron coincidence and multiplicity counter setup, operation, and performance. Several neutron multiplicity counters (NMC) were used by the participants in the laboratory after a lecture based review of the basic principles of passive neutron coincidence and multiplicity counting. Students went through the steps required to set up and operate a neutron multiplicity counter. First the student measured and became familiar with several detector characteristics including the deadtime, efficiency, die-away time, and gate fractions. Using a set of standards the students calibrated the counter for passive coincidence counting. Several special items with different matrices were measured to demonstrate the effect that matrix type has on the different analysis techniques. The students performed assays on many different types of items to become familiar with data analysis techniques and data correction procedures along with the advantages and disadvantages of the different analysis techniques.

## 3.3. Calorimetry NDA

The calorimetry part of the INSS NDA workshop lasted one day and included both a laboratory exercise and several lectures (Figure 6). The main goals of the training on this day was to describe the fundamentals of performing a thermal power measurement for a variety of different calorimeter designs, demonstrate how calorimetric assay combines isotopic information with the thermal power measurement from calorimetry to determine the mass of plutonium, illustrate the implementation of a measurement program for calorimetry and to provide the students with a hands-on experience in measuring SNM with a calorimeter.

Figure 6. Calorimetry Training at LANL

The initial laboratory exercise familiarized the students with the various components of a calorimeter, demonstrated the MultiCal software which operates the calorimeters, and began calorimetry measurements on two Plutonium items. The two calorimeters that were used to perform these measurements were a Wheatstone twin-bridge calorimeter which operated in passive mode, and a Wheatstone gradient bridge calorimeter which operated in servo or power replacement mode.

While the calorimetry measurements were approaching thermal equilibrium in the laboratory, a series of lectures were then given on safeguards calorimetry. The participants learned about the high accuracy and precision of calorimetric assay of plutonium relative to other NDA techniques. Focus was also given on the calorimetry measurements themselves, with an emphasis placed on the capabilities and typical modes of operations of a number of different types of calorimeters that are currently in use in the DOE complex and around the world. Comparisons were made between the performance and capabilities of Wheatstone bridge and solid-state thermopile calorimeters. The various uses of calorimetric assay within the DOE complex were also discussed, as well as its use to create secondary or working standards for other NDA techniques. In addition to reviewing the basic concepts of calorimetry, participants were also exposed to advanced concepts such as the various environmental and design factors that may affect the performance of a given calorimeter as well as techniques to decrease the measurement time that is needed to perform a calorimetry measurement.

The final session of the day involved the participants collecting the results from the calorimetry measurements, and determining the mass of the Plutonium items that were measured based on the calorimetry results and the isotopic measurements that were performed on these items earlier in the workshop. The day finished with a lively discussion of the calorimetric assay results.

## 4. Holdup Workshop at ORNL

The participants traveled to ORNL for extensive hands-on training on process holdup measurements on October 25–27 (Figure 7) and to attend the International Workshop on Best Practices for Material Holdup Monitoring, sponsored by the INMM held at ORNL from October 29 through November 1, 2006.

The ORNL workshop focused on practical exercises related to holdup measurements using the Holdup Measurement System, Version 4 (HMS-4) hardware and software. The exercise used seven fixtures to simulate pipe and duct assemblies in a generic processing facility.


Figure 7. NDA Workshop at ORNL

One of the goals was to demonstrate the capability to automatically perform hold up measurements for large process systems. Although holdup measurements can be tedious, complex and labor intensive, they are necessary to properly address closing the nuclear material balance for process systems. Therefore, it is important to demonstrate how automated measurement systems can be integrated into a site material control and accounting (MC&A) program to improve the capacity of the site to detect loss, theft, or attempted diversion of nuclear material. In addition, if properly implemented, a holdup measurement program provides commercial benefits to the facility because it can be used to identify the deposition of valuable nuclear materials. Once identified, these materials can be recovered.

Three systems were provided by the technical staff at the ORNL Safeguards Laboratory. The Chinese NDA delegation actively participated in the calibration of each of the measurement systems, using traceable NDA holdup standards. The systems were calibrated using the generalized geometry methodology. The Chinese NDA team was able to generate calibration constants that are used by the HMS-4 software to calculate the $^{235}U$ holdup for the various simulated process items located in the laboratory. In addition, the lectures provided information on specific details for the simulated process items. This information is necessary to store parameters for automation of the holdup measurement approach.

The Chinese NDA team also participated in the collection of measurement data for various simulated process items (Figure 8). Known quantities of uranium holdup were placed into a single pipe, a pipe array, an area source that simulated surface contamination and ductwork. The Chinese delegation was subdivided into three measurement teams. Each team conducted holdup measurements using the HMS-4 systems calibrated previously. During the measurement process, the U.S. technical experts provided input on the method and approach used by the Chinese teams. This included information on detector placement during background measurements, detector placement for foreground measurements, contributions from surrounding items, detector offset, and limitations for the measurement approach. The collected data were analyzed using the HMS-4 software program. Details of the analysis algorithms were discussed. The measurement results for each team were compared to the known quantity for each of the simulated process items.

Figure 8. Holdup exercises at ORNL

In addition to the holdup measurement exercise, the Chinese NDA delegation also participated in the INMM International Workshop. This workshop included presentations on best practices for holdup monitoring from subject matter experts from around the world. The Chinese NDA delegation was able to participate in the development of a document of best practices realized from the experience of subject matter experts that have been conducting holdup measurements for many years. This activity was important because it provided the Chinese with knowledge to improve holdup measurement techniques for civil process facilities in China that make a positive contribution to the overall domestic nuclear material control and accountability efforts.

## 5. Technical Tour to the Paducah Gaseous Diffusion Plant

The workshop concluded with a technical tour of the Paducah Gaseous Diffusion Plant (Figure 9) operated by the United States Enrichment Corporation (USEC) in Paducah, Kentucky, on November 2. The objective of this visit was to demonstrate how a site NDA measurement program is used to support MC&A activities at the site level and its contribution to regulatory compliance for a U.S. civil nuclear facility. The USEC staff was very supportive of the visit.


Figure 9. Technical Tour to the Paducah Gaseous Diffusion Plant

Operations management provided an overview of facility operations conducted in accordance with United States Nuclear Regulatory Commission requirements. Managers from the Analytical Laboratory then provided details on NDA measurements conducted by the site NDA group. These presentations covered topics associated with uranium enrichment measurements on $UF_6$ cylinders received on site, passive neutron measurements to determine the mass of uranium deposits formed in the enrichment cascade, and passive gamma measurements for waste containers. After the presentations, the Chinese delegation participated in a well-organized and

controlled tour of the cylinder lot, the enrichment cascade, and the NDA laboratory. The Chinese NDA group was able to observe enrichment measurements on $UF_6$ cylinders in the cylinder lot. In addition, the delegation was able to observe how procedures are used to ensure that data are collected in a quality manner and that results generated in the field are performed in accordance with administrative controls. After observing the enrichment measurements, the NDA team entered one of the process buildings to observe passive neutron measurements of piping in the cascade. The Chinese NDA team observed USEC personnel as they conducted gamma-ray scans of cascade piping to locate a uranium deposit, followed by passive neutron measurements to quantify the uranium mass for the deposit that was located using the gamma-ray scanning technique.

In addition, USEC personnel also demonstrated passive gamma-ray measurement techniques to quantify the mass of uranium in containerized waste. A collimated high-purity germanium detector coupled to a multichannel analyzer and laptop computer was used to collect data on the container. USEC personnel described site-approved methods calibrating the measurement system. They also provided an overview of the measurement approach and analysis software that is used to estimate the uranium mass for the container. The Chinese NDA delegation was very observant and actively participated in the technical discussions.

## 6. Conclusion

The seven Chinese delegates who participated filled out attendee questionnaires after each workshop at LANL and ORNL as well as the technical tour to the Paducah gaseous diffusion enrichment plant. The majority of the delegates who had more than three years of experience in the field found the learning experience beneficial, the course met their needs and was relevant to their job. The instructors were praised by all for their knowledge of the course content, effectively communicated technical details of the course material, and hard work. In addition, the practical exercises were very useful to the Chinese delegation to better understand the technical complexities for conducting NDA measurements. The overall response to their participation in the INMM-sponsored International Workshop on Best Practices for Material Holdup Monitoring was also positive but tempered by the fact that they did not have the opportunity to participate in the entire workshop because of travel constraints. The delegates were thankful for the effort involved in putting on the course and social events.

Based on feedback from instructors and the evaluation forms completed by the workshop participants, it can be concluded that the INSS NDA workshops were well organized and conducted successfully, the workshop goals and objectives were achieved and very well received by the participants.

## ACKNOWLEDGMENTS

This work was supported by the U.S. DOE NNSA's Office of Global Security Engagement and Cooperation (NA-242) and the CAEA. The authors wish to acknowledge Christine Shannon (LLNL), Stephanie Vigil (LANL), and Shannon Morgan (ORNL) for their part in the technical and logistics support and for making the workshops run successfully and smoothly. We also wish to acknowledge managers and staff of CIAE, LANL, ORNL, and USEC for hosting the visit and giving the technical and culture tours during the workshop. Finally, many thanks to all workshop instructors of LANL (David Bracken, Mike Browne, William Geist, Peter Karpius, Peter Santi, Duc Vo, Tracy Wenz, Richard Williams) and ORNL (C.A. Gariazzo, Steve Smith) for their hard work and excellent job.

# Oak Ridge National Laboratory's Safeguards Laboratory and Educational Programs

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***Abstract:***

*Technology plays a major role in the implementation of nuclear material safeguards and security. Both domestic and international programs rely heavily on advanced instrumentation. Recognizing this, the International Safeguards program at the Oak Ridge National Laboratory (ORNL) has established a laboratory in which equipment can be tested in a controlled environment. ORNL's Safeguards Laboratory (SL) is a unique user facility devoted to hands-on testing, evaluation, validation, and training for integrated safeguards methods, procedures, and instrumentation. The SL features a wide variety of nondestructive assay (NDA) instrumentation, special nuclear material (SNM) standards, and staff with a wide range of expertise. Combined with easy access to U.S. citizens and foreign guests, these features allow the SL to position itself as an excellent educational and training facility.*

*Over the last several years, SL staff have developed curricula for several NDA courses, which have included such topics as NDA basics, uranium enrichment measurements, and uranium holdup measurements. In addition, SL hosted a number of customized classes for various organizations, including the Air Force Radiation Assessment Team and the Radiological Assistance Program Team.*

*Several U.S. universities have recently started to develop educational programs that are focused on nuclear nonproliferation and international security. As a part of these programs, SL will provide technical support and facilitate hands-on training in the implementation of safeguards systems and instrumentation. A pilot class, "Nuclear Fuel Cycle and Nuclear Materials Safeguards," was held during the Spring 2007 semester at the Nuclear Engineering Department at Texas A&M University. The SL staff conducted a laboratory session of the class, which provided students with practical knowledge and experience.*

**Keywords:** safeguards; laboratory; training; education

## 1. Introduction

Technology plays a major role in the implementation of nuclear material safeguards and security. Both domestic and international programs rely heavily on advanced instrumentation. Recognizing this, the International Safeguards program at the Oak Ridge National Laboratory (ORNL) has established a Safeguards Laboratory (SL) as a unique facility devoted to hands-on testing, evaluation, validation, and training for integrated safeguards methods, procedures, and instrumentation. The SL features a wide variety of nondestructive assay (NDA) instrumentation, special nuclear material (SNM) standards, and staff with a wide range of expertise. Combined with easy access to U.S. citizens and foreign guests, these features allow SL to position itself as an excellent nonproliferation education and training center. As such, the SL will be able to link research and prototype equipment development with commercial and industrial organizations and promote the development of in-field methodologies for the end users through after-market training using real-life scenarios.

## 2. Features

The SL features a diverse array of safeguards-related inventory, including detection and measurement instruments, tags, process equipment mock-up arrays, SNM, and other radioactive material sources. To support in-field research and applications, equipment available from the laboratory includes a variety of gamma-spectroscopy instruments (portable solid-state and scintillation detectors, multi-channel analysers), neutron counting systems (active well coincidence counter), and support instrumentation (thickness gauges, oscilloscopes, and bar code readers). A set of industrial equipment mock-ups allows for simulation of SNM holdup simulations. Source materials include a uranium certified reference material (CRM) set with an enrichment range of 0.31% to 93.17% $^{235}$U. The nuclear material is available in a variety of geometries and chemical forms.

SL staff have broad expertise and many years of experience, including specialists in the fields of international safeguards, nuclear engineering, health physics, and NDA. Adjunct support to the SL staff is available from an immense array of technical specialties through ORNL staff (engineers, researchers, and scientists and additional guest researchers who visit ORNL annually for use of the laboratory's unique facilities).

ORNL is in the final stages of constructing 13 new facilities that support missions related to nanophase materials, advanced microscopy, and supercomputing. Additionally, the newly completed Spallation Neutron Source, in combination with the recently upgraded High Flux Isotope Reactor, makes ORNL the world's foremost center for neutron research.

All of these vital capabilities are available to support the mission of the SL and its parent organization, the International Safeguards Program.

## 3. Activities at the Safeguards Laboratory

The ORNL SL is a user-oriented facility largely dedicated to research, testing, and training in the areas of nuclear nonproliferation and international safeguards. In recent years, SL activities have focused on developing, testing, and evaluating strengthened safeguards measures to be applied globally by the International Atomic Energy Agency (IAEA) and providing technical support in implementing safeguards at nuclear facilities in the United States and abroad. The SL is currently supporting DOE sponsored activities in 11 countries.

While the SL has a broad technical foundation through ORNL, its primary activities are concentrated on the development of new NDA measurement procedures and the evaluation of commercially available instrumentation for safeguards purposes.

### 3.1. Training Courses

A number of hands-on courses are conducted at the SL on a regular basis. The laboratory configuration allows for both classroom instructions and individual hands-on stations for students. Training curriculum includes instruction in the areas of NDA, safeguards, and health physics, and, more specifically, NDA fundamentals, uranium isotopic measurements, uranium holdup measurements, and others.

During 2006 and the beginning of 2007, 10 different courses were held at the SL. Target audiences included representatives of nuclear authorities of China, Iraq, Russia, DOE-NNSA Headquarters, and various facilities and agencies around the United States. A sample of courses taught at SL is presented in Table 1. Continuing Education Credits (CEC) which are required for periodic re-certification by Certified Health Physicists (CHPs) were granted for by the American Academy of Health Physics for one course which had CHPs in attendance. SL continues to seek opportunities to gain CEC credit for these and other areas of professional certification or license.

| Course Name | Target Audience |
|---|---|
| Holdup Measurement Workshop | Electrochemical Plant, Zelenogorsk, Russian Federation |
| Holdup Measurements: Hands-on Approach | Personnel from various U.S. facilities and laboratories |
| Holdup Measurement Workshop | China Atomic Energy Authority |
| INMM International Workshop on Best Practices for Material Holdup Monitoring | International and domestic |
| Rapid Radioactive Source Identification Training | Air Force Radiation Assessment Team (AFRAT) |
| Fundamentals of Gamma Spectroscopy | National Guard Civilian Support Team |
| NDA Primer | DOE NNSA International Nuclear Safeguards and Engagement Program |

**Table 1:** Recent Training Courses Conducted at the ORNL SL.

### 3.2. Collaboration with Universities

Recently, several universities have started educational programs in the field of nuclear nonproliferation. As a part of the ORNL's university outreach program and in an effort to support new university programs, the SL is providing resources and expertise for hands-on laboratory exercises.

As a pilot course, SL hosted a laboratory session of the Texas A&M University course "*Nuclear Fuel Cycles and Nuclear Material Safeguards*" during the week of March 5–9, 2007. Professors William Charlton and David Boyle and 11 graduate students participated in safeguards instrumentation demonstrations and gained hands-on practical experience in nondestructive quantitative and isotopic analysis of SNM.


**Figure 1**: Laboratory Exercises at SL.

Topics covered during the class included uranium isotopic measurements, uranium holdup measurements, in situ object characterization software, and portable NDA instrumentation operation. As a part of the class, students attended three technical tours: the ORNL Environmental Effects Laboratory, the ORNL Radiation Effects Test Facility, and the Germanium Crystal Growing Facility at Canberra Industries, Inc.

This event was one of SL's first experiences in establishing closer relations with universities. All the visiting students and professors departed with very positive impressions and offered constructive feedback about the class. They were excited to have the opportunity to use advanced NDA equipment and to have access to SNM that otherwise is not available to those in a university environment. After the course, interest was expressed in continuing this effort by conducting similar events/courses several times each year.

### 3.3. Student Activities

By participating in ORNL's educational internship programs, the ORNL SL is continuously hosting students from various universities around the United States as research assistants or interns. At the SL, these students get a chance to work on research and training projects using state-of-the-art equipment and to interact with prominent subject matter experts.

During the last two years, four students have had the opportunity to work on research projects at the SL. Their research projects resulted in work that was presented at ORNL-wide student poster sessions, eight peer-reviewed papers presented at international conferences, and one ORNL Technical Report. In addition to working on research projects, the students actively participate in training courses as lecturers and instructors for hands-on exercises.

### 3.4. Cooperation with Vendors

Private companies such as AMETEK/ORTEC, Nucsafe, Canberra Inc., ICx Radiation, and others design and manufacture precision detectors, signal processing electronics, software, and systems for industry, academia, and government requirements. Their products include equipment and instruments for nuclear power plants and government nuclear facility operations, SNM safeguards, and fundamental research. The ORNL SL makes resources available which are not otherwise accessible in the private sector by supporting technology transfers and economic development. Thus, companies like these often request access to certain sealed radioactive standards, such as SNM. The standards are necessary for testing radiation detector systems by collecting spectra using isotopes that the instruments are designed to detect and characterize but which are not available in the manufacturer's inventory.

The unique capabilities, resources, and easy access to the SL make it attractive for commercial vendors of nuclear measurement instrumentation for testing and developing new instrumentation. Over the history of the laboratory, several commercial vendor activities took place at the SL. In 2005, Target Instruments[1] and Dr. Ray Gunnink[2] performed tests of NaIGEM for portable instrumentation (Target identiFINDER). Another example of recent visits includes one by AMETEC/ORTEC to conduct performance evaluation of a prototype "fission meter" neutron detection instrument. Several activities with ICx Radiation and Canberra Industries are planned for the near future.

### 3.5. Research Activities

One of the primary missions of the SL facility is research. SL's main research projects focus on testing, evaluating, existing, and developing innovative safeguards measurements techniques and procedures.

One recent project investigated the applicability of a mechanically cooled high-purity germanium (HPGe) portable detector for in situ process holdup measurements. Until recently, mechanically cooled HPGe detectors did not provide sufficient resolution for this type of measurement because of vibrations. Recently, a new portable high-resolution HPGe detector spectroscopy system with a

[1] Target Instruments is currently ICx Radiation.
[2] Dr.Ray Gunnink is a world-known author of uranium and plutonium isotopic measurement software packages like MGAU and NaIGEM.

miniature Stirling-cycle cooler (the trans-SPEC from ORTEC) became available. The use of a new long-life mechanical cooling system and noise reduction algorithms significantly improved detector resolution and eliminated the need for liquid nitrogen. The applicability of this new instrument for isotopic and quantitative analysis of uranium was verified by SL staff. Two software packages were chosen to test the trans-SPEC's performance: PC/FRAM (Fixed-Energy Response function Analysis with Multiple Efficiency) and ORTEC ISOTOPIC. PC/FRAM was used for uranium isotopic analysis and ISOTOPIC for quantification analysis. PC/FRAM has strict gain-setting and energy-calibration requirements, which are different from the "standard" factory parameters for trans-SPEC and the ones that are usually used by ISOTOPIC. Performance of ISOTOPIC with a set of PC/FRAM parameters was investigated and shown to provide better results than the standard set of parameters. Using PC/FRAM results for the normalization of $^{238}$U quantities also contributed to the overall better precision of the results [1].

Another project involves joint collaborations between the U.S. Department of Energy/National Nuclear Security Administration (DOE/NNSA) and the Brazilian-Argentine Agency for Accounting and Control of Nuclear Materials (ABACC), who agreed to cooperate in testing and evaluating isotopic enrichment measurement software. The two agencies are investigating the effects of inconsistencies in the types of hardware and types and versions of software used to determine the isotopic abundances for uranium and plutonium. This project is significant because it provides a regional forum for technical experts to evaluate and standardize isotopic enrichment measurement software used to verify operator declarations for enriched uranium. DOE/NNSA and ABACC initiated the project by establishing the Isotopics Measurements Working Group (IMWG), which is testing existing software packages and providing guidance for standardizing and upgrading the software to broaden the applicability, improve the data quality, and ensure that software quality assurance objectives are being met.

Another SL project is providing support to the ORNL and the European Commission Joint Research Center (JRC) at Ispra, Italy, by investigating the capabilities of and the applications for the Three-Dimensional Design Information Verification (3D-DIV) System at U.S. DOE facilities. The work was performed under a technical cooperation agreement between DOE and the European Atomic Energy Community (EURATOM). JRC developed the 3D laser scanning system for design information verification that is currently being used by the IAEA. The system is capable of creating 3D maps of rooms (and objects) and identifying changes in positions with a precision on the order of millimeters. The hardware and software were given to the SL to conduct capability tests under various conditions.

## 4. Conclusions

ORNL's Safeguards Laboratory was created as a facility focused on developing, testing, and evaluating strengthened safeguards measures to be applied globally by the IAEA and on providing technical support in implementing safeguards at nuclear facilities in the United States. and abroad.

Since the current lab was opened in October 2005, U.S. and international participants have attended numerous courses at the SL. The SL is also actively supporting ORNL's student and universities outreach program. Students are introduced to world-class research environments at ORNL and play a major role in research project activities. Recently, several U.S. universities have started to develop educational programs that are focused on nuclear nonproliferation and international security. As a part of these programs, the SL will provide technical support and facilitate hands-on training of safeguards systems and instrumentation.

Currently, there is an effort to designate ORNL's SL as a National User Facility. This designation would allow for an expansion of cooperation with universities, commercial vendors, and organizations around the world.

## 5. Legal matters

### 5.1. Privacy regulations and protection of personal data

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# Session 27

# Plants (measurement and instrumentation)

# Current instrumentation for physical verification in nuclear material safeguards as used by DG TREN

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## *Abstract:*

The European Commission has recently prepared a document on 'Implementing Euratom Treaty Safeguards', discussed and acknowledged with the EU member states. The document confirms that for credible nuclear material control physical verifications during inspections plays a major role. This paper presents an overview over instrumentation currently employed by the services of the European Commission for the different requirements of a variety of inspection strategies. The paper discusses non destructive assay and destructive assay methods, recent progresses, and underlines the importance of modelling for specific cases. It also discusses trends and directions where development is required. Synergy effects due to common use of equipment between the Commission and IAEA are discussed.

**Keywords:** instrumentation, measurements, physical verification

## 1. Introduction

The two largest international non-proliferation and nuclear material control systems both celebrate their fiftieth anniversaries this year. The treaty founding the European Atomic Energy Community (Euratom) was signed in Rome on 25 March 1957 by Belgium, France, Germany, Italy, Luxembourg, and the Netherlands - the IAEA statute entered into force on 29 July 1957 and the Non Proliferation Treaty on 5 March 1970.
Today, in 2007, 27 countries form the European Union and are thus parties to Euratom. The EU member states are a significant subset of the 188 (2005) states party to the NPT – and contribute about one third of the budget of the IAEA.

To satisfy the requirements of the Euratom treaty, the European Commission's Directorate for Energy and Transport (DG TREN) has built up a strong verification regime. The first inspections were run in 1960, today the implementation of the treaty provisions on nuclear material control are ensured by 180 accredited inspectors.

Today, the EC is in the process to modernize and adapt inspection strategies and approaches – most recently described in a paper on the *Implementation of Euratom Treaty Safeguards* developed in consultation with the EU Member States. Physical verification of nuclear materials and verification of the declarations is and remains a pillar of credible safeguards besides the accountancy of the nuclear material and performance controls through audits.

The following discusses instrumentation used by DG TREN for the various conditions under which physical verification of nuclear material needs to be carried out. It provides an idea of the status of instrumentation and verification techniques, recent and ongoing developments, and notes a few cases

where improvements are required. As DG TREN is running about one third of its inspections in the EU together with the IAEA, common use of equipment by EU and IAEA inspectors are discussed as well.

## 2. Physical verification as inspection task

The level of verification of the nuclear material depends on a number of variables. The safeguards approach for a certain type of facility, the historic and recent experience with the individual facility, the type of inspection, the available resources together with detection probabilities and goals enter into the equation.

Verifications can be done at very different levels of escalation, sophistication, effort – and not least cost. An inspector may simply count a number of containers, or compare their identities and completeness with the declarations. He may need to determine weights or characteristic attributes to determine material types. He may want to verify the amounts of declared material in containers with a low or very high precision. Or he may be faced with the challenge of verifying the inventory or throughput of a complex bulk handling facility (e.g. a reprocessing plant or a mixed oxide fuel fabrication factory). For all these situations the appropriate tools are required.
Challenges to the technical support services providing these tools are manifold – they should have the required accuracy, be lightweight for portable equipment, be reliable, be simple to use etc. and the various instruments should have a limited number of different interfaces, to standardize as much as possible. This is not only required for economic reasons but is also a key for reliable use by an inspector, who is typically charged with a large variety of different tasks and cannot be required to be an instrumentation or measurement specialist. If the equipment is installed in a facility, plant boundary conditions need to be taken into account and reliability and data authenticity are but two of the additional issues.

## 3. Portable and installed equipment for physical verification

Equipment for physical verification is often categorised in two groups – portable equipment and installed instrumentation. Another distinction is between non-destructive assay (NDA) and destructive assay (DA) methods. Most of the portable equipment is NDA related and used more frequently in item facilities. An exception are the small and (trans-) portable Compucea laboratories, used to analyse samples in LEU fuel fabrication plants during physical inventory verifications.

Standard portable NDA equipment of DG TREN are for example hand held gamma spectrometers and active or passive neutron coincidence counters (NCC). Whereas the gamma spectrometers are mainly applied as attribute testers, to verify enrichment, or properties of spent fuel, NCCs in various designs are used to determine the mass of Pu or U containing items.

Installed equipment is usually found in bulk handling facilities processing very large amounts of strategically very significant materials. Clearly, these plants have a high significance for inspections. Instrumentation for bulk handling facilities is often integrated into the processes. This is due to the facts that the industrial processes are often fully automated, access for inspectors to material is difficult and standard non-destructive assay (NDA) instrumentation is not applicable. The major facilities where the Commission has invested into installed verification equipment are the mixed oxide fuel fabrication plants in the United Kingdom and France as well as the reprocessing plants in these two countries. New facilities under construction concern large storages of direct use material.

Whereas the plant integrated NDA tools provide partial defect information and contribute to containment and surveillance/ monitoring of nuclear material, credible verification requires bias defect control as well. Thus samples need to be taken and analysed with high accuracy. The level of accuracy which is required can still only be provided by destructive assay (DA) methods. At the sites of the reprocessing facilities, where the number of samples is particularly large, on site laboratories have thus been installed by the European Commission.

### 3.1 Reprocessing facilities

Four large-scale reprocessing facilities are under Euratom Safeguards in the UK and France. Subdivided into storage of spent fuel, dissolution and chemical separation plants and product storages, each part of the facilities is equipped with appropriate verification equipment, described e.g. in (1).

*The chemical plants- flow verification*: Recently, a significant modernization effort has been started. Tank verification systems are being renewed and in particular the data analysis of the signals derived from tank transfers is being enhanced and improved, see (2). This will also lead to a higher degree of technical standardization for the two sites as far as the software used for verification is concerned.

*The chemical plants – quantitative analysis*: Quantitative information on the material input and output is mainly derived from the accountancy tanks after dissolution of the spent fuel, from DA samples and weighing of the Pu nitrate solutions after separation, and from input measurements to the product stores, where the focus is on quantitative Pu-oxide verification with NDA techniques.

A major area of modernization concerns here the gamma spectroscopy. As the burn-up of the reprocessed fuel is increasing, the isotopic vector of the Pu is changing, slowly reaching the limits of the currently applied techniques. Investigations on second cycle fuel (Pu extracted from spent MOX fuel) have been triggered and are ongoing. Many of the issues encountered in this field have been discussed at a dedicated workshop (3) which has re-launched interest in development in this field.

The Pu-oxide stores of the reprocessing facilities are equipped with fully automated NDA systems, acquiring data of the complete material flow. Data acquisition and analysis is carried out with the Commission developed software package RADAR/ CRISP (4). The modernization from a variety of older systems to this standard solution is almost completed.

### 3.2 On site laboratories

Key for quantitative verification at the reprocessing facilities is the destructive analysis of samples. As the number of samples is very large, as transport is expensive and due to the safety risks heavily regulated, and as inspectors need verification results in a timely manner to use them in a meaningful way, the European Commission decided to implement on site laboratories at the sites of the reprocessing facilities at La Hague (LSS) and Sellafield (OSL).

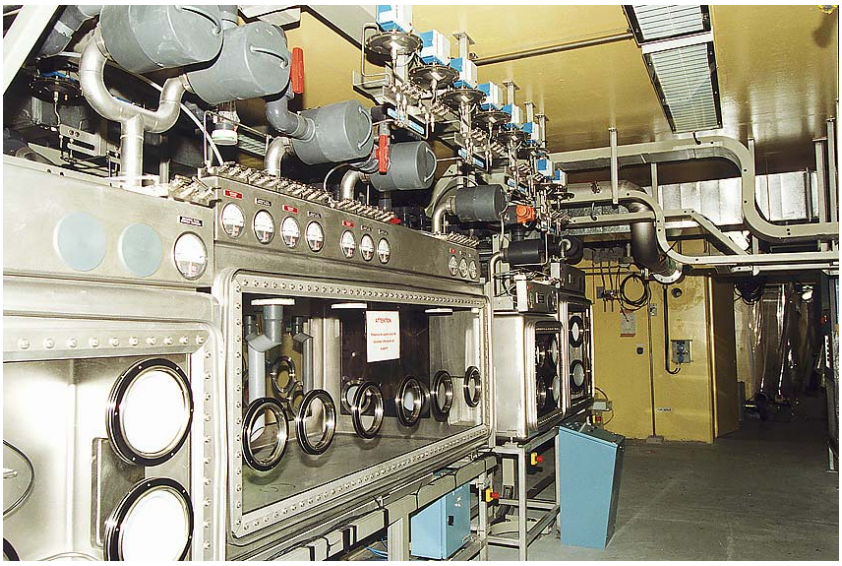

*Fig 1 View on one of the glove boxes of one of the on site laboratories of DG TREN*

They have been operational now since 1999 and 2000 respectively. The number of analysed samples is reaching almost one thousand per year. The laboratories are equipped with glove boxes containing instrumentation ranging from sample handling robots for sample preparation, over high resolution gamma spectrometers, neutron coincidence counters to K-edge spectrometers, and mass spectrometers etc (5). Their analysis capacity is matched to the inspectors accountancy verification requirements for the four reprocessing plants on the two sites. At Sellafield, the OSL also analyses samples taken at the Sellafield MOX plant, SMP.
The laboratories are owned by DG TREN and operated by the Institute of Transuranium Elements, ITU, of the Joint Research Center of the European Commission, who maintains a team of analysts dedicated to work at the laboratories. Despite the unusual boundary conditions for an analytical laboratory, the sample throughput and particularly the attained level of accuracy are excellent (6). This is e.g. documented in the result of an external quality control programme, comparing different laboratories and focusing on Uranium assay. The data points marked OSL refer to NDA(left) and isotope dilutes mass spectrometry results (IDMS, right).

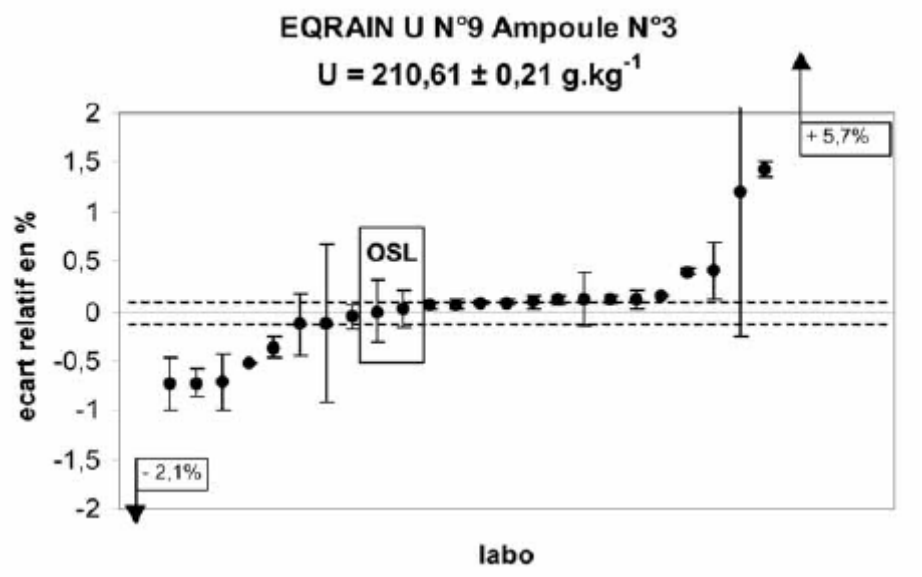


*Fig 2 Results of external quality control programme including OSL data for NDA and IDMS (from ref 6)*

The cost of the laboratories is significant and has always to be balanced with the benefit for inspections.

Modernization of the laboratories is to be envisaged for the near future, as particularly the mass spectrometers at the OSL are reaching their end of life.

The continuing success of the laboratories, running since 1999, has been an important factor for the IAEAs decision to set up a similar laboratory at the Rokkasho-Mura reprocessing facility in Japan. This activity is supported by the European Commission's Support Program to the IAEA, mainly by direct support from ITU, including training of IAEA staff at the Commission laboratories.

### 3.3 Uranium fuel fabrication facilities - DA: Compucea – 2nd generation and beyond

At LEU fuel fabrication facilities, samples are taken throughout the year and particularly during the annual physical inventory verification (PIV).

In order to provide timely high accuracy verification results at these installations during the PIV, the instrument Compucea (Combined Procedure for Uranium Concentration and Enrichment Assay) has been developed several years ago. The portable laboratory has had significant success and, run by specialized ITU staff, reaches excellent performance The results are usually made available to the inspectors (EC and IAEA) before the end of the week in which the PIV is carried out.

Increasing difficulties to transport sources across Europe, and issues with the acquisition and disposal of radioactive sources have led to a very successful new development.

The second generation Compucea is entering its production phase now (8)! The main change is the move from K-edge spectrometry to L-edge spectrometry for the concentration measurement. The radioactive sources used for the latter could thus be replaced by an X-ray generator. This results in a significant reduction in organizational effort for transports and avoids the need to regularly acquire new sources and decommission old ones.

The next step, another very promising development is already ongoing (10): Herbert Ottmar reports in this conference on his test to prepare the replacement of Ge detectors by LaBr3 scintillators. This would further simplify the Compucea set up by avoiding the use of liquid nitrogen and making the instrument even smaller, lighter and easier to transport.

It is to be underlined that so far all improvements have been achieved by ITU without compromising the measurement performance or accuracy.

### 3.4 Recent improvements and developments of NDA equipment

In the area of non destructive measurement equipment, a big standardization effort was made around the year 2000. This has led to a fairly stable instrumentation park, with which inspectors are familiar and which covers the routine requirements, particularly as far as portable equipment is concerned.

The focus of this chapter will thus be on some specific new developments which could be recorded recently.

Non destructive assay workhorses for DG TREN remain gamma spectrometers and neutron coincidence counters.

DG TREN owns about 200 gamma spectrometers – hand held devices (identiFINDER/HM-5 (9), EM-1) or portable acquisition chains (MiniMCA with NaI/CZT/HPGe detectors (10)) both for attribute testing or enrichment measurements. In addition, a significant number of gamma spectrometers is installed in large facilities, forming part of automatic verification systems.

For the mass determination of U and Pu, neutron measurements of various kinds are employed. Passive HLNCC detectors for Pu-oxide canisters, active well coincidence counters for high enriched Uranium and active/passive neutron collars for LEU or MOX assemblies, pins or small samples have been standard for many years. DG TREN owns around 50 of these devices, which is a significant share of those existing world-wide.

A large number of neutron coincidence counters is installed in bulk handling facilities handling direct use material. For these units, the improvement of Monte Carlo modeling techniques has been extremely useful. In most cases, they need to be modeled individually. It is also typical that they cannot be calibrated with completely independent standards prepared elsewhere, e.g. certified reference standards. Again, Monte Carlo modeling is very useful in these cases and has been employed quite successfully.

One example is a neutron coincidence counter which is integrated into a magazine handler of a mixed oxide fuel fabrication plant (11). A more recent second example

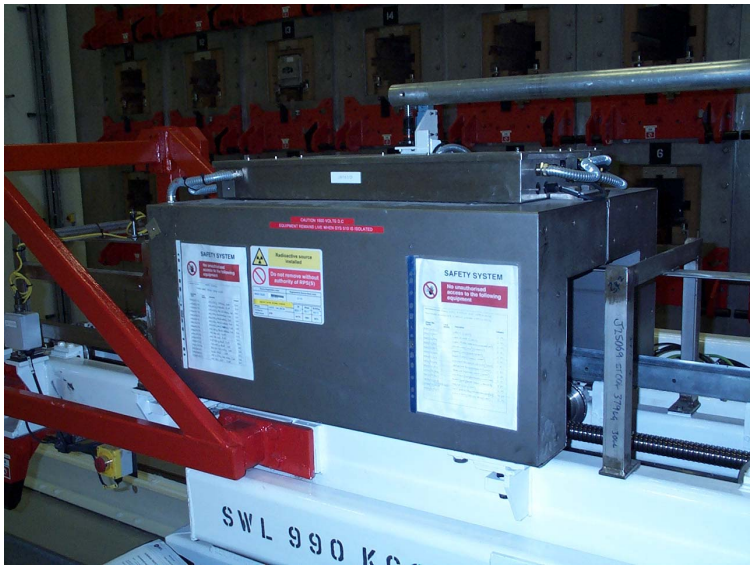


*Fig 3: Neutron coincidence counter integrated into magazine handler (from ref 11)*

concerns a detector which will be used to verify Pu-oxide containers after their import into a mixed-oxide fuel fabrication plant. This detector needs to be integrated into an existing and running plant after a review of the safeguards strategy. The position where this detector needs to be located provides only very limited space. Without optimization of the design by Monte Carlo methods it would have been far more difficult and risky to come to a satisfying solution (12).

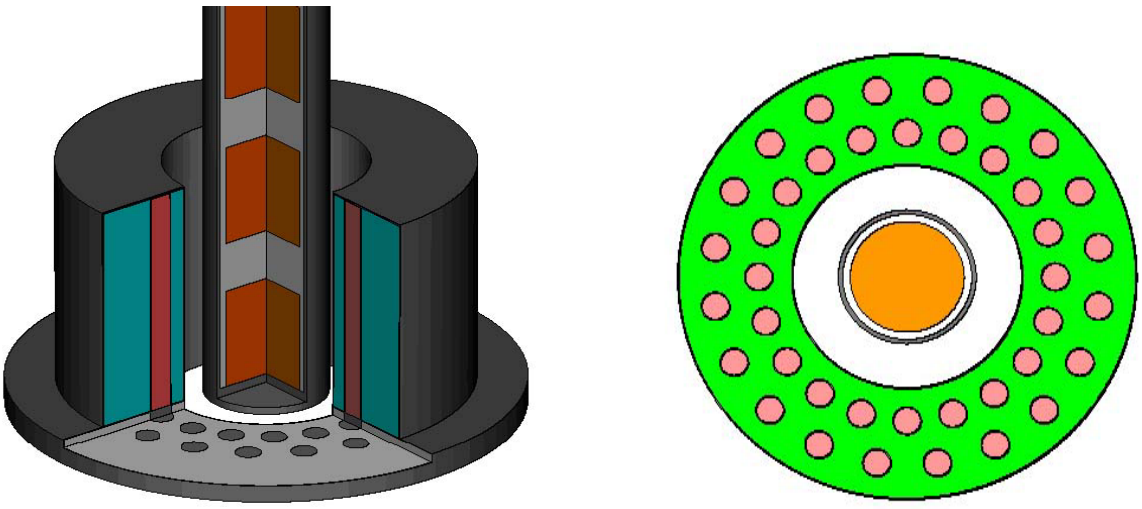

*Fig 4: design of a neutron coincidence counter for Pu-oxide verification (from 12)*

Other specific developments for unusual situations that have been carried out recently include the optimization and very successful calculational calibration of two detectors used for the verification of fuel assemblies for highly enriched uranium assemblies used at the research reactor of the University of Munich (13) and the High Efficiency Passive Counter (HEPC), a very large neutron coincidence counter which is being used for the verification of LEU powder containers in two fuel enrichment facilities (14).

### 3.4.1 New tool for under water MOX measurements: UWANEGA

A new instrument has recently been developed to for the purpose of attribute testing of fresh MOX material under water, the UWANEGA (Under WAter NEutron and GAmma). In a particular plant situation, the use of the Underwater Coincidence Counter, which is normally used for partial defect tests on MOX under water, was not feasible due to technical constraints, where MOX pins were stored in closed containers.

An instrument has thus been designed, which is fairly easy to use from the bridge of a fuel pond handling machine. One 3He tube and one collimated CZT detector are mounted inside a steel tube. Cabling is fed through steel tubes to the bridge, where the data acquisition is done with standard electronics (MiniMCA). Meant as simple attribute tester, the instrument actually performs much better. The gamma spectrometer shows Pu spectra as expected. The neutron data, however, show fairly linear behaviour between totals count rate and declared 240Pu mass. This was a particular surprise as the geometry is not perfectly well defined during the measurements (15). A more detailed report on UWANEGA will follow.

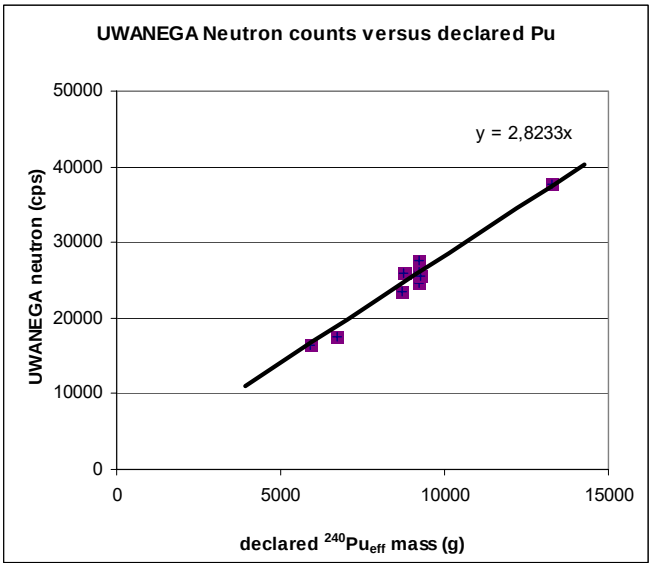


Fig 5 UWANEGA detectors and relation of total neutron counts vs declared 240Pu mass

### 3.4.2 Current issues for NDA instrumentation

The above has shown that for standard and many very specific cases DG TREN disposes of adequate and excellent instrumentation to aid the inspectors in their task of physical verification of nuclear material. It has been a considerable success that it was possible to standardize on a few building blocks for a variety of also quite specific and dedicated measurement tools, so that the technical complexity of the design leaves only little or no trace at the interface which the user needs to handle.

One lesson learnt with standardization is that it takes time and that time is an antagonist to standardization. The technical development moves on and the tools which have been developed in the last years need to be replaced.

Particular challenges in this respect are to be found in the measurement chains of both NDA workhorses – gamma spectrometry and neutron coincidence measurements.

The neutron shift register JSR12 is, by design, more than 15 years old and a replacement is due. A point is approaching, where the inspectorates will both have to decide whether they want to continue with new electronics of the same line, which is expected to appear this year on the market, or whether the acquisition of list mode data is a viable alternative. Taking the decision, the value of a standardized solution should be kept in mind, albeit only as one of the parameters.

On the gamma spectroscopy side, the MiniMCA has been introduced at DG TREN, then Euratom Safeguards Office, in 1997. The instrument hardly ever fails but the key components disappear from the market and a successor needs to be selected in the foreseeable future.

## 4. Cooperation with the IAEA

The above demonstrates the capacity of DG TREN to provide to Commission inspectors the required tools for physical verification. Inspections in the non nuclear weapons states of the European Union are carried out together with the IAEA under the verification agreement INFCIRC 193 (Infcirc 193 is in force in 17 of the 25 NNWS, for the remaining eight entry into force is in preparation). In these inspections, it is normally avoided to duplicate efforts of the inspectorates. For the common inspections, the instrumentation is agreed beforehand. Installed equipment is normally purchased and maintained by one organization but commonly used, sharing the total burden and cost. Formally a List of Agreed Instruments is maintained, which Liaison Committees between the two organizations agree.

For both inspectorates, considerable synergy effects can be mobilized if the cooperation is well implemented. Throughout their co-existence as nuclear material inspectorates inspecting Europe, the IAEA and the European Commission have co-operated in many development projects, frequently also with the involvement of the Euratom or member state support programs to the IAEA. Positive recent examples are the development of the MiniMCA, the optimization and calibration of the detector for FRMII fuel or the integration of an enrichment algorithm into the IdentiFINDER. The Commission's support program activities with respect to the Rokkasho on site laboratory or the Compucea development are but two more very important examples from the area of measurements. DG TREN is prepared to enhance the cooperation with the IAEA for mutual benefit.

## 5. Conclusion

DG TREN is well prepared to meet the requirements for physical verification of nuclear material as requested and described in the new implementation paper for European safeguards. DG TREN disposes of a large variety of equipment and has a significant and well trained pool of inspectors to make best use of them in inspecting nuclear installations. The infrastructure to undergo new developments is well developed and the cooperation between DG TREN and the JRC provides for a solid scientific base, which also provides an excellent foundation for safeguards in Europe under the non proliferation treaty as carried out by the IAEA jointly with DG TREN.

### 6. Acknowledgements

The authors would like to acknowledge the contribution of many colleagues, particularly those of the JRC whose contribution was and is essential and whose work is quoted frequently, but also those who are not named here.

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# Safeguards Technology Development for Advanced Fuel Cycles

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***Abstract***

*The expansion of nuclear energy around the world, particularly as relates to advanced fuel cycles, presents both opportunity and challenges from a safeguards and nonproliferation viewpoint. New facilities and processes present the opportunity to enhance the nonproliferation regime by strategically incorporating advanced safeguards into the design from the very beginning. The primary challenge is to effectively implement safeguards as the throughput of these new facilities increases, making it all but impossible to achieve detection goals in a timely manner using traditional nuclear material accounting measurements alone (nondestructive and destructive). In this paper we examine the safeguards technology needs for advanced fuel cycle technologies, and develop the concept of the 'safeguards envelope.' Areas of additional research and development in current state-of-the-art measurement techniques (particularly neutron and gamma-ray based nondestructive assay) are identified as well as novel new approaches such as super-high resolution gamma spectrometry using micro-calorimetry, laser induced breakdown spectroscopy, and UV/VIS spectroscopy. Finally, we outline the integration that will be required to achieve full utilization of the individual components of the safeguards system to achieve near real time knowledge of facility operations.*

**Keywords:**: safeguards systems, NDA instrumentation, neutron, gamma, reprocessing

## 1. Introduction

Global electricity demand is projected to more than double by 2050, with nuclear power comprising a significant and growing share [1,2]. At the same time, the disposition of spent nuclear fuel remains an issue, with some countries reaching the limit of their current pool storage capacity and consequently having to seek alternatives such as dry storage. As new nuclear power plants are built in the future, it is clear that waste management, including minimization, will become ever more important. Expansion of nuclear power also brings with it potential proliferation concerns. Advanced fuel cycle concepts have recently been proposed and explored as a way to address one or more of these issues. As a result, the safeguards community has begun to evaluate challenges and opportunities associated with them from a nuclear materials management perspective.

The United States recently announced a new nuclear energy initiative, the Global Nuclear Energy Partnership (GNEP), which introduces an advanced fuel cycle concept that addresses increasing energy demand, minimizes volume and heat load of spent nuclear fuel, and employs both intrinsic and extrinsic measures to address proliferation issues [2].

Under GNEP, plutonium and the minor actinides from spent nuclear fuel (SNF) are recycled as fuels for an advanced fast spectrum reactor. The recycling process employs group actinide recovery to avoid having separated plutonium, enhancing the intrinsic barrier to misuse. Extrinsic measures of the GNEP concept include the development of advanced safeguards technologies and systems to optimize the effectiveness of domestic and international safeguards [2].

In this paper, we outline research and technology development ongoing and needed to address the safeguards challenges posed by GNEP and other advanced fuel cycle concepts.

## 2. Safeguards challenges posed by advanced fuel cycles

One of the fundamental challenges to safeguards presented by advanced fuel cycle concepts is in the intrinsic material properties present throughout the entire fuel cycle. Concern over separated plutonium accumulation has driven these concepts to utilize enhanced radiation as a barrier to misuse. While this is a benefit from the perspective of hindering access, these same properties make quantitative measurement more difficult. For example, group actinide separation will result in a mixture of plutonium and minor actinides, including curium, which will in turn dominate the passive neutron emission and thereby impact standard neutron measurement approaches. Some fission products also remain in the final fuel product, resulting in a high gamma-ray dose and making standard isotopic measurements more difficult.

Associated with intrinsic materials properties is a practical challenge to safeguards, namely the extensive use of hot cells throughout the recycling and fuel fabrication process. This translates to equipment that must operate reliably in a much harsher environment. Not only will instrumentation need to be robust in a high radiation environment, maintenance schemes will be needed to accommodate the restricted access associated with such facilities.

Facility throughput represents another challenge. As throughput increases, the IAEA goal of 8 kg Pu for the detection of protracted and abrupt diversion represents an ever increasingly smaller fraction of the total and at some point additional measures must be taken to supplement standard nuclear material measurements. For facilities with throughputs on the order of 1000 ton heavy metal or more, the 8 kg Pu goal represents less than 0.1% of the total. On the other end of the spectrum, there are challenges for small throughput facilities in the case where safeguards detection goals are based on a percentage of the active inventory. This is the case for both NRC and DOE licensed facilities in the U.S., where the detection goals are 0.1% and 1% of the active inventory respectively.

Pyroprocessing technology is being evaluated as a recycling option and presents a special case as there is not an input accountability tank with which to establish initial inventory as there is for aqueous processing. The relative non-homogenous nature of this process presents a particular challenge to chemical analyses which rely on small samples.

Finally, expansion of nuclear power will result in greater transportation of nuclear materials. This represents a challenge for maintenance of continuity of knowledge and for shipper-receiver differences.

## 3. Technology development needed to address challenges

Addressing these challenges requires advances in instrumentation, systems analysis and modeling, and data integration and knowledge extraction, but also provides an opportunity to evaluate the application of safeguards in an integral sense and to develop a ‘defense in depth’ approach [3]. The opportunity also exists for building in safeguards requirements into the design process, thereby maximizing their efficacy and optimizing the associated costs. This ‘safeguards by design’ approach is being employed for the envisioned U.S. GNEP facilities.

One of the basic challenges that can be addressed by advances in instrumentation falls under the general category of the direct measurement of spent nuclear fuel. This is obviously the case for input to the recycling process, and remains the case as a result of the intrinsic radiation barrier present throughout the fuel cycle. As a result, even final fuel assemblies produced will present difficulties for measurement that are similar to input SNF. A recent study has identified seven technologies in various states of maturity that have potential to improve the capability to measure SNF [4]. These range from active to passive neutron methods, employing both prompt and delayed signatures. Given the variety of sample types to be encountered, it is likely that more than one technology will be required.

The fact that neutrons from curium are ubiquitous in GNEP and other advanced fuel cycle concepts leads to the potential of expanding the neutron balance approach throughout to address not only plutonium mass determination but also for continuity of knowledge between the reactor, storage, and recycling facilities [5].

Instrumentation based on x- and gamma-ray measurements are also a stable of the existing standard suite of nondestructive assay methods and will need to be adapted to accommodate the enhanced radiation properties of advanced fuel cycle materials. Recent advances in microcalorimetry have demonstrated an improvement in energy resolution over the best high purity germanium detectors by an order of magnitude in the 100-keV region [6]. This technology can also be applied as an alpha spectrometer, again achieving much greater resolution than current detection methods and could facilitate destructive analysis by pre-screening samples.

Evaluation of the basic nuclear physics data is needed to accompany instrumentation development. Current efforts involve evaluation of the status of spontaneous and induced fission multiplicity distributions, delayed neutron and gamma-rays, and (alpha,n) yields. In addition to basic data, uncertainties are needed to enable sensitivity analyses to be performed. For example, what additional information can be extracted using ratios of higher neutron multiplicity moments and is the existing data precise enough to be useful? Nuclear physics data is also critical for modeling and simulation used for source term analysis, detector design, and general radiation transport within a facility. Data for minor actinides will be increasingly important for multi-pass recycling scenarios.

Online monitoring is another area where research and technology development can play an enabling role. Building instrumentation directly into the process can not only enhance real time knowledge of facility operations, it can also help to ease the burden of sample taking. Development of both radiation and non-radiation based methods for online deployment can provide process monitoring information that can be used to augment traditional safeguards measurements. When combined with other data, a general picture of the operation of the facility can be developed using an explicit model approach [7]. Non-radiation based instrumentation, such as ultraviolet/visible spectroscopy (UV/VIS) and laser induced breakdown spectroscopy (LIBS) may provide information that complements that obtained from radiation based sensors. Determination of ratios of minor actinides to plutonium should be possible with the LIBS technique and such technology has been applied at THORP. Both UV/VIS and LIBS could be employed to monitor chemistry changes which may be indicative of facility misuse.

Systems analysis and modeling play an increasingly important role as the integration of individual safeguards measures is achieved, as well as enabling a holistic evaluation of facilities. Capability is needed for optimization of specific safeguards designs as well as evaluation of performance in real time. A critical aspect of this is information technology in all its manifestations, from basic data flow and organization to encryption and data security.

Putting it all together is the concept of the ‘safeguards envelope’ where data from traditional safeguards, process monitoring, containment and surveillance, personnel movements, etc, is folded together to form a confidence measure that a facility is operating normally [8]. By utilizing all available data, one can envision parameterization in such a way that not only are confidence intervals developed for individual components of the system, but also for aggregates thereby accounting for correlations between disparate data. In addition, experience with such a system could lead to indicators that are more predictive as opposed to reactive in nature, much like observation-based preventative maintenance in non-nuclear industries.

Basic technology advances have common application in both domestic and international safeguards. In order to meet these challenges in a systematic way, the GNEP program has establish a domestic safeguards campaign, which will provide much of the basic research and technology development needed to implement advanced safeguards. This effort will be coordinated with and complementary to the developments undertaken by NNSA for international safeguards.

## 4. Summary

The global increase in the utilization of nuclear energy, particularly in the form of advanced fuel cycles such as GNEP, present both challenges and opportunities for safeguards and nonproliferation. A holistic approach to meeting these challenges through a dedicated research and technology development program offers the potential to achieve significant advancement in nuclear materials management on a global basis. Through the GNEP domestic safeguards campaign, a focused and prioritized effort is being

undertaken to address these challenges for the proposed U.S. facilities. Results of these efforts will provide enabling technology for both domestic and international safeguards.

## 5. Acknowledgements

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# A Next-Generation Automated Holdup Measurement System (HMS-5)

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***Abstract***

*Since its release in 2004, Holdup Measurement System 4 software (HMS4) has been in use at facilities to systematically measure and verify the amounts of uranium holdup in process facilities under safeguards. It is a system used for measuring uranium and plutonium and archiving holdup data (via bar-coded locations with integrated information), a capability which is essential to any internationally safeguarded facility's efforts to monitor residual special nuclear material (SNM). The HMS4 has been tested by sites in Russia, the United States, South Africa, and China, and based on feedback and lessons learned received thus far, an updated version of the software would enable international partners to use a broader range of commercial equipment existing at these facilities.*

*In June 2005, the Oak Ridge National Laboratory (ORNL) and Los Alamos National Laboratory conducted a holdup measurement training course on HMS4 for subject matter experts from the Ulba Metallurgical Facility at Ust-Kamenogorsk, Kazakhstan. An external software package was used that improved measurement of low-enriched uranium by adding higher-energy gamma rays to the interpretation protocol. This dual-energy approach, which is currently not integrated into HMS4, would greatly benefit the next-generation HMS software package (HMS-5). This software upgrade would also provide the International Atomic Energy Agency (IAEA) with a more comprehensive software package for testing at several safeguarded locations.*

*When released, HMS4 only supported AMETEK/ORTEC and GBS Elektronik equipment, despite the use of Canberra Industries technology [detectors, multi channel analyzers (MCAs), other hardware, and software packages] at many facilities. For HMS-5 to support more hardware systems and to benefit the majority of international partners and the IAEA, it must accommodate Canberra technology because of such widespread use of its hardware. Furthermore, newly developed hardware such as lanthanum halide detectors, mechanically cooled germanium detectors, and digital signal-processing multichannel analyzers will be incorporated into the new HMS-5 system to accommodate evolving SNM detection and quantification technology. HMS-5 is a natural extension of automated special nuclear material holdup measurement systems. ORNL is leading the development of this next-generation system with the assistance of its foreign partners and its experienced Safeguards Laboratory staff.*

**Keywords:** safeguards; holdup; material accountancy; software; foreign collaboration

## 1. Introduction

Holdup material is the residual amount of special nuclear material (SNM) remaining in a processing facility after the bulk materials have been cleaned out. This material has eluded material accountants for years when trying to determine the exact amount of "material unaccounted for" (MUF) in SNM processing facilities. A way to measure the amount of unknown material in any given geometry was developed by scientists at the Los Alamos National Laboratory (LANL). The Generalized Geometry Holdup (GGH) analysis methodology was formulated to simplify the quantitative analysis by measurements performed using portable gamma-ray spectroscopy systems. Major factors that must

be taken into account in interpreting holdup measurements include the unique geometry of each individual holdup deposit and its associated background signature.

An automated system was devised to take holdup measurements of SNM in various types of facilities. This system slowly evolved into the industry-standard Holdup Measurement System 4 (HMS4) – developed as a joint effort between LANL, Oak Ridge National Laboratory (ORNL), and the Y-12 National Security Complex and licensed by AMETEK/ORTEC. HMS4 has been in use at facilities to systematically measure and verify the amounts of uranium holdup in process facilities under safeguards since its release in 2004. It is a system for measuring uranium and plutonium and for archiving holdup data, which is essential in order for any internationally safeguarded facility to monitor all amounts of residual uranium and/or plutonium (specifically MUF). Archival efforts are facilitated by the use of bar-coded locations with integrated information automatically read into the software. Additionally, HMS4 has been extensively tested by sites in Kazakhstan, Russia, the United States, South Africa, and China for more effective application.

## 2. History of Automated Holdup Measurement

The original HMS was an early holdup assessment concept used for ensuring compliance with criticality safety limits. It was not a quantitative system. The next series of holdup measurement systems (HMS2, HMS3, and HMS4) are software packages for performing and documenting quantitative SNM holdup measurements. Below is a brief history of the various generations of the holdup measurement system software package.

### 2.1. Holdup Measurement System 2

The first version of the automated holdup measurement system software was created in 1992. This was the Microsoft FoxPro program, Holdup Measurement System 2 (HMS2). It was designed and developed as a cooperative effort by LANL and Oak Ridge Y-12 National Security Complex. After testing the software, raw data were collected and used to validate the code. The data were collected in the well-defined experimental setups normally used for the annual U.S. Department of Energy (DOE)–sponsored LANL Holdup School. Manually calculated holdup values were compared with the values determined by the software by LANL Safeguards Sciences Group (NIS-5) staff members for each experimental setup to verify the HMS2 software calculations. Subsequently, the software was validated, accepted, and used as a part of the class.

### 2.2. Holdup Measurement System 3

In 1997, the next version, Holdup Measurement System 3 (HMS3) v1.0, was developed. Microsoft (MS) Visual Basic 4 was used for HMS3 design to transfer HMS2 into the MS Windows operating environment. To validate and verify this new software package, the original raw data collected in the 1992 HMS2 verification exercise were used and re-formatted for HMS3. Analytical results from HMS3 agreed with previous results using HMS2. ORNL and LANL accepted the validation of the software, and HMS3 was then used as part of the LANL holdup class. This software package was licensed to and sold commercially by ORTEC. In 1999, the HMS3 (v1.0) software was updated to version 2.0 to comply with the MS Windows 32-bit technology. MS Visual Basic 6 was used to recompile the new version. The same verification method was used to validate the version 2.0 software.

### 2.3. Holdup Measurement System 4

Development of Holdup Measurement System 4 (HMS4) was started in 2001. This was another cooperative project between LANL, Y-12, and ORNL. The project was sponsored by the Safeguards and Security Division of the DOE Safeguards Office and managed by LANL. The design of the new version was based on the original HMS3 approach and implemented and automated many new holdup measurement methodologies developed by LANL. The approach used for the validation and verification of HMS4 was the same as that for HMS2 and HMS3.

#### 2.3.1. HMS4 Features

HMS4 is a MS Windows–based software package that has evolved from earlier program versions. It is written in MS Visual Basic .NET® as part of the MS Visual Studio .NET® 2003 development package and uses MS Access® (MS Office 2000/XP format) database files. The reports are generated with the Crystal Decisions, Inc., Crystal Reports banded report generator, which is included with the Visual Basic .NET package. Software for the controllers (Pocket PC devices) is written in MS eMbedded Visual Basic® as part of the MS eMbedded Visual Tools v3.0® development package for Windows CE. Windows-based HMS4 offers the user a menu-based environment.

HMS4 contains several important holdup correction algorithms such as *finite source correction* and *self-attenuation correction*. These algorithms help account for geometric factors that affect final results. HMS4 supports 20 spectral regions of interest (ROI) to aid users measuring plutonium. The data from each measurement period (or measurement campaign) are easily accessed from the main menu. All measurement data dumps are date and time stamped and allow for an 80-character comment field which can be used for extra notes. Serial communications between the host PC or the bar-code reader/controllers and the various MCAs utilize DLLs (dynamic link libraries) designed by the MCA vendors. Several improvements also have been made in the way that HMS4 performs background calculations.

HMS4 includes two sets of programs: the main program that runs on a host personal computer (PC), and the other that runs on a bar-code reader or portable PC. The bar-code reader or portable PC is often referred to as the controller. The main host computer program performs setup and calibration of MCA/detector pairings, loads the controller(s) with operational parameters, receives measurement data from the controller(s), maintains measurements and derived results in databases, and prints reports. The field controller program controls MCA setup functions and data acquisitions, stores accumulated measurement data, and allows the user to review previously collected data and spectra. Four host PC programs are included in the HMS4 package: (1) the stand-alone Windows-based controller program (HMS4 Controller), (2) a stand-alone MCA control selection program (MCA Switch 2), (3) an upgrade program (HMS4mdb) to be used for updating existing, older HMS3 databases to the HMS4 format, (4) and a program (Spectra Split) for extracting embedded ROI information from a controller-saved spectrum.

HMS4 supports four types of MCAs: the (Ametek/Ortec) DART, the (Ametek/Ortec) MicroNOMAD, the (Ametek/Ortec) DigiDART, and the (GBS Elektronik) Rosendorff MCA-166. The field controller software of HMS4 also supports several bar-code readers such as the 700 series reader by Intermec Corporation and the 8000 series by Symbol Technologies. Both of these reader series use the Microsoft Pocket PC 2003 operating system. HMS4 also supports state-of-the-art detectors. The detector types supported by HMS4 must be a gamma-ray detector appropriate for the MCA chosen: a sodium iodide (NaI) detector, a cadmium-zinc-telluride (CZT) detector, or a high-resolution Germanium (HPGe) detector. [1]

### 2.3.2. Current Status of HMS4 Implementation

Currently, HMS4 is being utilized for uranium and plutonium holdup measurements in various countries around the world including DOE facilities in the United States such as the K-25 Uranium Enrichment Plant (Bechtel Jacobs Company, LLC.) outside of Oak Ridge, Tennessee. New capabilities being developed at LANL and ORNL are being incorporated into training workshops such as the LANL-developed dual energy method for analyzing low-enriched uranium (LEU) deposits in addition to measuring quantities of highly enriched uranium (HEU). These training courses have been given in China and the United States as well as to nondestructive assay (NDA) scientists from Russia, South Africa, Kazakhstan, and the United States. Furthermore, university programs in the United States have taken advantage of the accessibility of ORNL's Safeguards Laboratory to provide hands-on HMS4 experience and SNM holdup measurement training for their students by the ORNL Safeguards Laboratory staff.

At ORNL's Safeguards Laboratory, training courses have been conducted for personnel from various U.S. facilities and laboratories as well as from private industries (e.g., Canberra and ORTEC). Currently, HMS4 is being evaluated for use at the Y-12 National Security Complex and the Idaho National Laboratory and has been used at various LANL facilities and at the Rocky Flats Plant outside Denver, Colorado. (The Rocky Flats Plant used HMS3, the previous generation system.)

## 3. Holdup Measurement System 5 (HMS-5)

Since the release of HMS4, several new methodologies were developed and new generations of hardware systems have become available. The use of HMS4 over the past few years has also helped identify certain limitations and shortcomings of the software. The following sections discuss those limitations associated with the current holdup measurement system and introduce some improvements planned for the next version: Holdup Measurement System 5 (HMS-5).

### 3.1. Limitations of HMS4

Some of the currently identified limitations of HMS4 since its release in 2004 have been outlined below. This is not an all-inclusive list, as other issues are being identified with continued use of the system.

1. Increased measurement uncertainties for quantification of uranium below 20% enrichment
2. Lack of popular hardware/software integration (Canberra)
3. Compatibility with only four MCAs
4. Difficult adding hardware/software modulations
5. Complex summarizing master reports interface
6. Complex method for updating algorithms
7. No support for foreign-manufactured MCAs (except GBS Elektronik Rosendorff MCA-166)

### 3.2. Planned Improvements in HMS-5

Improvements planned for HMS-5 to address these limitations in HMS4 include incorporation of an automated method of analyzing the quantity of LEU using the LANL-developed dual energy approach. Currently, this method is limited to inputting data into and calculating results from a spreadsheet. The desire is to incorporate this analysis automatically into the system's operation so that HMS-5 is not limited to certain types of processing facilities.

HMS4 primarily uses the 186-keV gamma ray for uranium holdup measurements because it is a direct measure of $^{235}U$ and because it has a moderate yield (~43,000 gammas/second/gram $^{235}U$). Self-shielding of this low-energy photon can be significant in thick deposits, and it would be beneficial to use the 1001-keV gamma ray from $^{238}U$ to assist in quantifying uranium in LEU holdup.

Canberra hardware and software were not accommodated in the initial HMS releases. This omission is to be corrected with the full cooperation of Canberra Industries to include the Inspector-2000 MCA, other Canberra detectors, and Canberra software such as Genie2000. Furthermore, since its release, new hardware and software package versions have been released by a majority of manufacturers and other third-party suppliers. These updates must be accommodated in HMS-5. For example, ORTEC's Maestro has gone through many versions since 2004 and new drivers/support are needed to accommodate such changes. Additionally, HMS4 is dependent on Microsoft software packages, which are constantly being updated: MS Access® (database files) and MS ActiveSync (for synchronizing the field controller with the host PC).

Other features intended for the next-generation holdup measurement system include separating various calculations and constituents of HMS4 into individual modules to facilitate updating certain aspects of the program (drivers, algorithms, database information, etc.) and localizing the software (converting the software to be used in various foreign countries, such as translating menu languages). Furthermore, current users of HMS4 have expressed the need for a more user-friendly interface with the system that relies more on graphical representation. A graphical user interface (GUI) is planned for HMS-5 to simplify operations for users (ranging from facility operators to inspectors) and to facilitate localized versions of the system for use in other countries (Russia, Kazakhstan, and China have expressed interest).

Lastly, as with any gamma spectroscopy system, updates must be made for newly developed equipment and hardware. Currently, HMS4 is able to run with NaI, CZT, and HPGe detectors. Lanthanum halide detectors (specifically, lanthanum bromide) have recently been introduced and

have medium resolution (comparable to CZT) yet are as robust, portable, and efficient as the NaI detectors (a significant advantage for making field measurements in harsh environments). Alternatively cooled HPGe's also have been introduced by ORTEC and Canberra. Aside from the added benefit that these mechanically cooled detectors have the characteristically high resolution of HPGe's, they require no liquid nitrogen (LN) for cooling the crystal during spectral acquisition and hence are much more portable than conventional LN-cooled HPGe's. Higher resolution detectors identify more peaks, which allows more extensive calculations and yields better self-attenuation correction approximations.

Other planned features of HMS-5:

- Correction for the presence $^{212}$Pb 238-keV photopeak
- Installation of authorization procedure and user rights
- Selection of particular runs (points) within date dump for summary master
- Improved dialogue for criticality safety concern measurement points
- Background correction with the presence of equipment (attenuation by equipment itself)
- Support of foreign hardware (e.g., Russian Greenstar MCAs)
- Use of 1-MeV gamma photopeak (from $^{238}$U daughter decay) to identify infinite thickness in uranium deposits

### 3.3. Development with International Partners

Since its release in 2004, HMS4 has been taught to NDA scientists from many countries: Kazakhstan, Russia, China, South Africa, and the United States. As a joint effort between scientists from LANL, ORNL, IAEA, and the Ulba Metallurgical Plant (UMP) of Kazakhstan, HMS4 was used for determining quantities of LEU holdup as well as HEU holdup in ventilation and vacuum systems in fuel fabrication facilities. A training course was given at LANL on the Auto Duel Energy Analysis method for holdup measurements in June 2005, and an advanced training course was given in May 2007 at UMP in Ust-Kamenogorsk, Kazakhstan. The Auto Duel-Energy Analysis method for measuring LEU quantities is one of the main new developments to be considered for inclusion in HMS-5. UMP is serving as a test bed for incorporating this method and is providing invaluable insight into the measurement campaigns needed for this analysis. [2]


**Figure 1:** UMP Holdup Training Course at LANL, June 2005.

In October 2005, NDA personnel from the Electrochemical Plant of Zelenogorsk, Russia, were trained in the use of HMS4 for measuring and reporting uranium quantities. The Russian site receives HEU

oxide produced from the decommissioning of nuclear weapons. The site converts the HEU oxide to $UF_6$ and down blends the HEU with LEU to produce a $UF_6$ product suitable for conversion to reactor fuel. When the site receives their equipment next year, two HMS4 systems will be used; one in the fluorination facility and the other in the sublimation facility. Currently, they are translating the software into Russian for more widespread regional use of HMS4 and exploring the possibility of adapting their version of HMS4 with the ORNL Safeguards Laboratory staff to make it compatible with Russian hardware (e.g., Greenstar MCAs).


**Figure 2:** Zelenogorsk Electrochemical Plant Holdup Training Course at ORNL SL, October 2005.

This benefits the development of HMS-5 by using the Zelenogorsk site's experience in taking measurements using other/foreign hardware not previously available to HMS4 developers. Furthermore, the Zelenogorsk site has agreed to modify the current version of HMS4 to best fit the needs of Russian SNM processing facilities by performing menu/language translation and obtaining certification by Russian authorities for expanded regional use.

**Figure 3:** Holdup Training Course for Chinese NDA Scientists at ORNL, October 2006

The ORNL Safeguards Laboratory is working with the Chinese Institute of Atomic Energy (CIAE) to enhance SNM holdup measurement capabilities in China. More than 36 Chinese participants from various organizations and facilities attended an NDA Workshop from October 9–13, 2006, at the CIAE in Beijing, China, given by ORNL Safeguards Laboratory staff and LANL NDA scientists. The workshop consisted of fundamental NDA measurement principles, how they are implemented on the nuclear fuel cycle, SNM holdup measurement theory, and training on using HMS4. Later that month, seven Chinese technical experts were selected to attend an extensive hands-on training course on process holdup measurements using fixtures to simulate pipe and duct assemblies in a generic uranium processing facility. This training course advanced cooperation between LANL, ORNL, and CIAE to further enhance SNM holdup measurement capabilities in China as well as provided the possibility to expand HMS-5 to an extremely broad audience.

Over the past 3 years, the South African Nuclear Energy Corporation (NECSA) has acquired three fully operational HMS4 systems and is looking to apply these systems to measuring bulk material in a uranium conversion facility. Recently, three NECSA NDA scientists have been fully trained on HMS4 operations and have expressed interest in the LEU measurement and analysis method for NECSA to use on a nation-wide scale in the future.

## 4. Conclusions

The aim of this project is to create the integrated technology that enables routine, plant-wide measurements of holdup for accountability and for safety at DOE facilities and other facilities around the world. It enhances the effectiveness of in situ measurements and improves facility operations by reducing the need for equipment cleanout for inventory verification. It also offers a standardized approach which helps reduce potential errors and measurement uncertainties. Commercialization with a U.S. manufacturer supports goals for industrial competitiveness and improved technology in U.S. industry and supports DOE collaborations with industry.

## 5. Acknowledgements

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# $UF_6$ Sampling Method using Alumina

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***Abstract:***

*The ABACC-Brazilian-Argentine Agency for Accounting and Control of Nuclear Materials has developed a $UF_6$ sampling method for enrichment determination (named ABACC-Cristallini Method) which uses a fluorothene P-10 tube type containing alumina pellets that absorb and hydrolyze $UF_6$ directly during the sampling. The alumina pellets retain up to few hundreds milligrams of U (in a solid compound – $UO_2F_2$) without the need of using liquid nitrogen during sampling. This new method has advantages compared to the actual method that uses a Hoke tube as: the $UF_6$ sample content left at the installation (archive sample) will be lower and less reactive, the laboratory procedures for manipulating the sample will be much easier, the residual uranium retained at the laboratory will be much lower, the sampling device is less expensive, there will be saves in transport cost as well, and it is relatively safer concerning radiological protection aspects during transportation.*
*This paper describes the physical principle of the new method, the experiments carried out at laboratory taking into account different process parameters foreseen in real cases, and the development of a procedure for recovering the uranium retained inside the alumina pellets for the U enrichment measurement. The behavior of the new method is compared to the traditional one, showing no loss of accuracy for the enrichment determination with real $UF_6$ samples taken from enrichment plants. The qualification strategy applied to the new method for routine safeguard application at the enrichment plants is presented in this paper as well.*

**Keywords:** *$UF_6$* sampling; enrichment plant safeguard; measurement techniques and standardisation

## 1. Introduction

The ABACC-Brazilian-Argentine Agency for Accounting and Control of Nuclear Materials performs safeguard inspections jointly with the IAEA-International Atomic Energy Agency at enrichment plants in Brazil and Argentina. Particularly, at enrichment plants in Brazil that use centrifuge enrichment process, routine and unannounced inspections are performed and $UF_6$ samples are taken from process lines and cylinders to verify the uranium enrichment conformity with design/operator declarations.

A Hoke tube type is normally used for $UF_6$ sampling. Sampling dwells up to one hour, with the tube immersed in liquid nitrogen, and up to 10 grams of $UF_6$ are collected. The samples taken in Brazil are sent by ABACC to a Network Laboratory in Argentina for mass spectrometry analysis. The $UF_6$ sample is hydrolyzed and a very small quantity (some milligrams) is used for the enrichment determination. The residual quantity of $UF_6$ retained at the laboratory is very large compared to the needs for the enrichment measurement. It is also a costly sampling system, as the Hoke tube type is expensive, it

has to be cleaned-up before reutilization, and additional costs are added to the transportation of cleaned tubes from Argentina to Brazil.

Due to the disadvantages of the actual $UF_6$ sampling method, ABACC has developed a method (named ABACC-Cristallini Method) of sampling $UF_6$ for enrichment determination. The new method uses a fluorothene P-10 tube type containing alumina pellets that absorb and hydrolyze $UF_6$ directly during the sampling. The alumina pellets retain up to few hundreds milligrams of U (in a solid compound – $UO_2F_2$) without the need of using liquid nitrogen during sampling. With this new method the $UF_6$ sample content left at the installation (archive sample) will be lower and less reactive as the actual, the laboratory procedures for manipulating the sample will be much easier (no need for hood, gas sampling, vacuum system, nitrogen cleaning, etc), the residual uranium retained at the laboratory will be much lower, the sampling device is less expensive, there will be saves in transport cost as well, and it is relatively safer concerning radiological protection aspects during transportation. Figure 1 shows the two types of $UF_6$ sampling device and the alumina pellets.

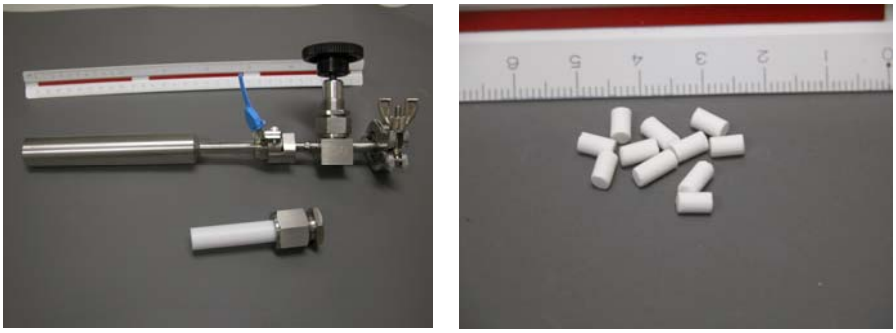

**Figure 1:** $UF_6$ sampling devices – Hoke tube / fluorothene P-10 tube and alumina pellets

## 1.1 Reaction Mechanism

Chemical traps are commonly employed in enrichment plants to remove $UF_6$ from various gas streams of the process. The effectiveness of the chemical trap largely depends on the particular absorbing material that is used to fill the trap. As these traps are the final barriers between the process and the environment, they should assure the total elimination of $UF_6$. The materials normally used in these traps are alumina ($Al_2O_3$) and sodium fluoride (NaF). The election of the appropriate absorber is made comparing different performance factors as: reaction kinetics, absorber regeneration / uranium recovery requirements, absorbing capacity, pressure losses, and effects of other system components such as HF and $F_2$ [1,2]. These factors do not have the same importance in every application. For the $UF_6$ sampling method proposed in this paper the predominant factors are reaction kinetics and uranium recovery.

The sodium fluoride trapping mechanism involves the reaction of $UF_6$ to form a solid complex, according to the following reaction:

$$2\ NaF + UF_6 \rightleftharpoons UF_6.2NaF$$

The uptake of uranium hexafluoride by sodium fluoride is a chemisorption process forming an $UF_6$.2NaF complex. This reaction can be readily reversed heating the complex to around 350°C and recovering the uranium as $UF_6$. Also, it can be dissolved in water, obtaining a solution with the uranium and a high NaF saline content. If the NaF, with the retained complex, is dissolved in water, the high fluoride quantity present in the resulting solution should be eliminated completely by successive evaporations, because its presence affects the isotopic analysis.

On the other hand, the effectiveness of the alumina relies on the hydrolysis of the $UF_6$ with the available lattice water. The reaction is the following:

$$6\ Al_2O_3.H_2O + 3\ UF_6 \longrightarrow 3\ UO_2F_2 + 6\ Al_2O_3 + 12\ HF$$

$$12\ HF + 2\ Al_2O_3 \longrightarrow 4\ AlF_3 + 6\ H_2O$$

The $UF_6$ alumina trapping mechanism depends on the hydrolysis reaction with subsequent retention of the uranyl fluoride ($UO_2F_2$) in the porous structure. The $UO_2F_2$ is a non-volatile solid and soluble in water. This facilitates the preparation of an adequate solution to carry out the isotopic analysis.

Due to the characteristics described above, alumina pellet was chosen as the material to be used for $UF_6$ sampling.

Some initial quantity of water is essential for the alumina loading mechanism, being a content of 3 to 4% near the optimum [1]. However, if the alumina, in the form of pellet, contains too much water, greater than 7 to 8%, it will be so reactive that the pores leading to the interior of the pellet will become prematurely plugged, and the uranium loading is largely confined to the periphery of the pellet. In this case, the average load can be considerably lower than otherwise expected. These facts shall be taken into account for the material specification and sampling setup.

### 1.2. Alumina Characteristics

The tested aluminum oxide is type gamma, used as catalyst support, bimodal, with a very high specific surface. The material has form of pellets of 1/8", (cylinders of 3 mm diameter and 5-6 mm high – see Figure 1) with an apparent density of 0.39 g/cm$^3$ and a total pore volume of 1.14 cm$^3$/g. The specific area is around 250 m$^2$/g measured by the BET Method.

The absorbed water was determined as 0.07% by heating at 120°C for 2 hours. The crystallization water was determined as 4.5% by heating at 1200°C for 2 hours. Considering the high specific area of the pellets, the material was not exposed to the atmosphere and maintained in its original and hermetic container in order to low the absorbed humidity.

To verify that the alumina did not contribute with any impurity, especially uranium, which could interfere or cause error in the determination of the U isotopic composition, a blank pellet was analyzed. Several washes with distilled water and $NO_3H$ 1M, as is applied to the recovery of the $UO_2F_2$, were carried out. The blank solutions were measured using the Total Reflection X Ray Fluorescence (TXRF) technique. The impurities detected by TXRF in the blank were the following:

- Iron: 0,15 µg/ml, equivalent to 1.8 µg/g in the alumina;
- Potassium and Calcium: 0.5 µg/ml, equivalent to 6 µg/g in the alumina;
- Uranium was not detected, being 0.02 µg/ml in the blank, equivalent to 0.2 µg/g in the alumina, the detection limit of the method.

Due to the presence of HF generated during the hydrolysis, the alumina presents certain solubility that increases with time and heating. Under the conditions selected for $UO_2F_2$ recovery, it was determined that the total alumina dissolution amounts to 0.05%. Such a low quantity does not cause any problem to the isotopic analysis by mass spectrometer.

## 2. Experimental setup

Figure 2 shows the equipment setup scheme used for the $UF_6$ sampling with alumina pellets at the Laboratório de Control Químico y Físico – Combustibles Nucleares – Centro Atómico Constituyentes – Comisión Nacional de Energía Atómica - Argentina.

The $UF_6$ cylinder and the fluorothene tube loaded with the alumina pellets are connected to the manifold. The loaded fluorothene tube, with its plug and nuts, are tare before its connection to the system in order to estimate the quantity of uranium retained in each experiment.

The manifold is dried out before starting each test by applying vacuum and heating (80-90°C) during 30 minutes. Then, the $UF_6$ cylinder valve is opened in order to obtain the desired pressure and valve 2 is opened to allow that the alumina pellets enter in contact with the $UF_6$. After having elapsed the elected time for the test, valve 1 and 2 are closed. To remove the $UF_6$ from the system, valve 3 is opened, being $UF_6$ retained by the cold trap with liquid nitrogen. Valve 2 is also opened to be sure that there are no detectable gases in the fluorothene tube. Valve 4 is then opened to allow the entrance of nitrogen gas for equalizing the internal pressure to the atmospheric pressure. Finally valve 2 is closed and the fluorothene tube is disconnected from the manifold. It is closed with the respective plug and

nuts and weighted to know the uranium mass retained in the alumina. In general, the procedure described above was applied on all experiments, with some variants according to the kind of test being done.

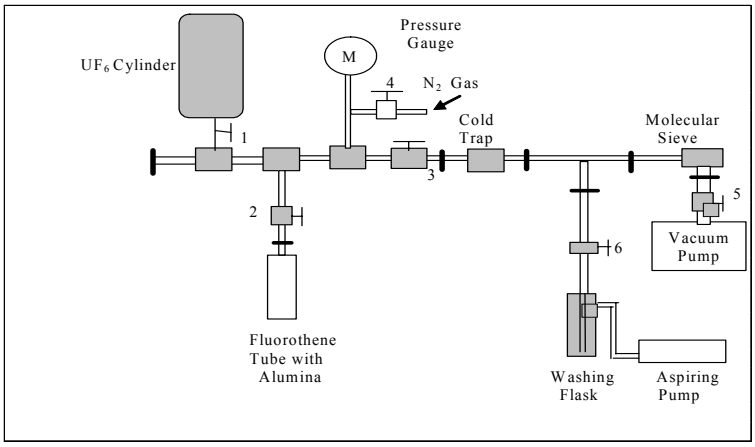


**Figure 2:** equipment setup for $UF_6$ sampling with alumina pellets

## 3. Experiments

Various experiments with fluorothene tube containing alumina pellets were carried out using $UF_6$ samples, which were obtained from ABACC´s inspections, having U enrichments between 0.31 to 4.05wt%. The first 5 experiments were carried out to set up the system. During these experiences some smaller losses took place, and the data was not reliable, but gave evidence that the uranium was retained by the alumina. Then the experiments were carried out at several constant pressures in the system to evaluate the quantity of uranium retained at a fixed time. Data were also collected when the $UF_6$ tubes were becoming empty, giving knowledge about the alumina behavior. In all the experiments, the fluorothene tube was loaded with 1g of alumina pellets without any previous treatment. Some representative experiments are described below.

### 3.1. Experiments carried out at decreasing pressure

Experiments with decreasing pressure in the experimental setup are presented in Table 1. Figure 3 shows the evolution of pressure versus time for one specific test.

| *Test Number* | *Alumina Mass (g)* | *Initial Pressure (mb)* | *Final Pressure (mb)* | *Contact Time (min)* | *Retained $UF_6$ Mass (mg)* | *Retained U Mass (mg)* | *U Mass / $Al_2O_3$ Mass Ratio* |
|---|---|---|---|---|---|---|---|
| #14 | 1.072 | 30 | 2.9 | 30 | 285 | 193 | 0.18 |
| #18 | 1.033 | 15 | 2.4 | 12 | 111 | 75 | 0.07 |
| #20 | 1.028 | 108 | 26.1 | 80 | 688 | 465 | 0.45 |

**Table 1:** experiments carried out at decreasing pressure

Test #18 showed a smaller U retention than the test #14, but the contact time was 12 minutes compared to 30 minutes in Test #14 and the pressure was also lower. Test #20 showed higher U retention for 80 minutes, but the final pressure of the system was 10 times higher than the other experiments because the alumina was coming closer to its loading limit.

According to Schultz [2] the maximum saturation load, expressed as the uranium to alumina mass ratio is between 0.4 and 0.5. The maximum values obtained in this work were something superior (up to 0.6 for all tests).

It is necessary to highlight that in Test #18, with an initial pressure as low as 15 mb, the pressure had a quickly decreasing because the tube of $UF_6$ was almost empty, but in only 12 minutes 75 mg of

uranium were retained by the alumina pellets. This quantity of uranium is more than enough to carry out an isotopic analysis. This is an indicative that it is possible to sample $UF_6$ from almost empty containers or plant pipes that have very low pressure.

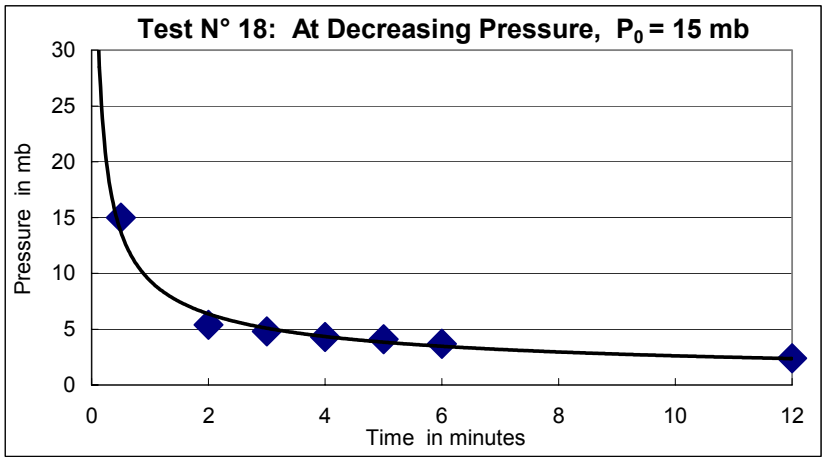


**Figure 3:** experiment at Decreasing Pressure

### 3.2. Experiments carried out at constant pressure

Several constant pressures in the system were set to evaluate the amount of uranium retained in a fixed time and the maximum uranium load for the alumina saturation as well. Table 2 presents some relevant data of the tests performed and Figures 4 and 5 show the pressure evolution as function of time. The pressure of the system was maintained as much as possible constant by regulating the $UF_6$ Hoke tube valve. After 60 minutes, the valve was closed but the pressure recording continued for 20 minutes more. It is clearly observed that the pressure keeps falling during the following 20 minutes in the case where the alumina is less loaded (Test #12, Figure 4), indicating a continuous $UF_6$ absorption. In opposite, Test #16 (Figure 5) shows that the alumina pellets seem to be practically saturated, with a very low decrease of the system pressure, and uranium to alumina mass ratio of 0.62.

It is evident that the reaction is quite quick despite of the several steps that are involved in this kind of gas-solid reactions:

- diffusion of the $UF_6$ molecule from the bulk gas stream to the external pellet surface;
- diffusion into the porous structure;
- adsorption on the interior surfaces;
- reaction with the absorbing material;
- diffusion of the $UF_6$ molecule through the complex layers to unreacted absorbers.

In Test #12 at a very low pressure of only 10 mb, within 60 minutes, 213 mg of uranium were retained. The fourth part of this U mass is more than enough to perform an isotopic analysis. Therefore, 15 to 30 minutes seems to be an appropriate time interval for sampling lines with $UF_6$ pressures among 10-100 mb, and using only 1gram of alumina pellets as specified by this method.

| *Test Number* | *Alumina Mass (g)* | *Initial Pressure (mb)* | *Final Pressure (mb)* | *Contact Time (min)* | *Retained $UF_6$ Mass (mg)* | *Retained U Mass (mg)* | *U Mass / $Al_2O_3$ Mass Ratio* |
|---|---|---|---|---|---|---|---|
| #12 | 1.031 | 10 | 3.1 | 60 + 20 | 315 | 213 | 0.21 |
| #17 | 1.067 | 25 | 17.1 | 60 + 20 | 662 | 447 | 0.42 |
| #13 | 1.018 | 50 | 42.3 | 60 + 20 | 773 | 523 | 0.51 |
| #16 | 1.040 | 100 | 95.3 | 60 + 20 | 947 | 640 | 0.62 |

**Table 2:** Experiments carried out at constant pressure

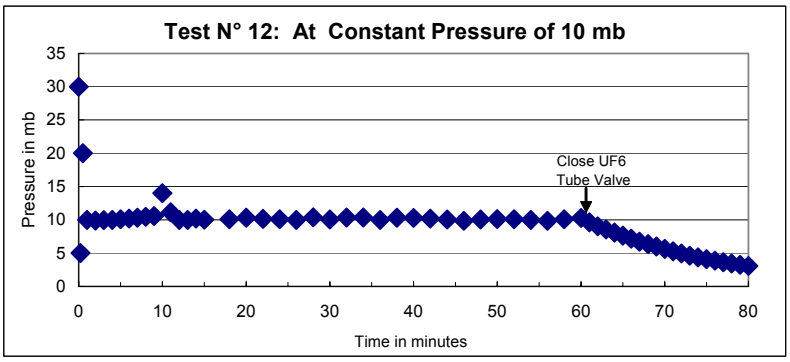


**Figure 4:** experiment at constant pressure test#12

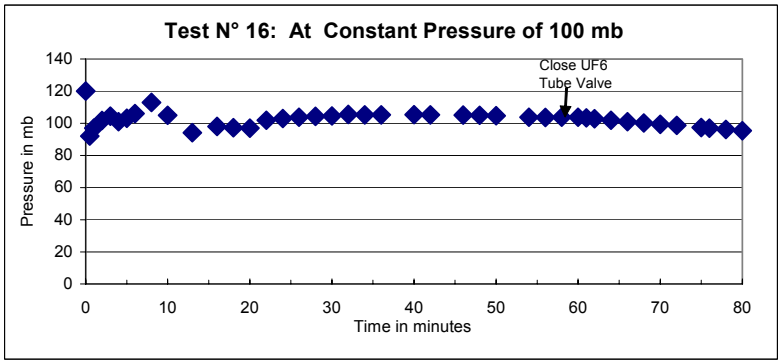


**Figure 5:** experiment at constant pressure test#16

### 3.3. Evaluation of the uranium mass retained as function of the system pressure

Figure 6 shows the uranium mass retained by 1 gram of alumina versus the $UF_6$ pressure inside the system. In this graph the value obtained at constant pressure tests were used (see Table 2). A similar curve can be obtained plotting the $U/Al_2O_3$ mass ratio versus the $UF_6$ pressure. The retained uranium was determined by the weight difference of the fluorothene tube loaded with the alumina pellets, before and after each test. Some data were also checked by measuring the uranium content by Davies & Gray Method.

The saturation value of $U/Al_2O_3$ mass ratio seems to be next to 0.62, which is the highest value obtained by Test #16, carried out at 100 mb of pressure during 80 minutes.
This is a high uranium retention value obtained with the alumina pellets used. This value is superior to the one obtained by Schulz [2], where values of 0.6-0.7 are consigned for the $UF_6/Al_2O_3$ mass ratio saturation, which are equivalent to 0.4-0.5 for the $U/Al_2O_3$ mass ratio.

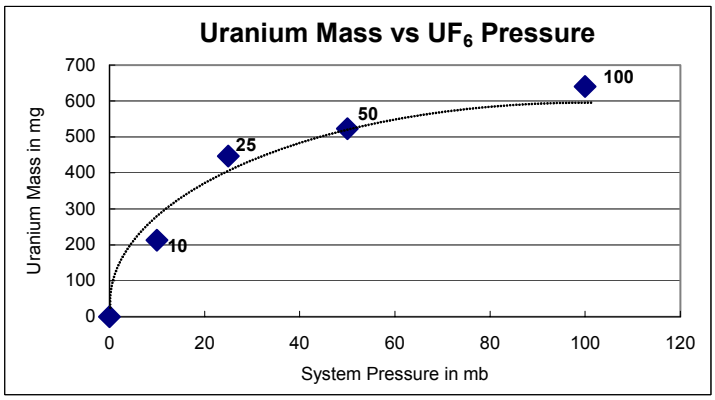


**Figure 6:** uranium mass retained in function of the system pressure

### 3.4 Recovery of the uranium retained in the alumina

As the uranyl fluoride ($UO_2F_2$) is a very soluble salt, the uranium retained in the alumina pellets can be removed with distilled water without any effort. However, due to the high material porosity, several washes with small water volumes and even some washes with 1M nitric acid are necessary for complete uranium dissolution. A high solution acidity, intense heating or prolonged contact times are not recommended, because the alumina is partially dissolved and the final solution will have a high aluminum content that shall affect the isotopic analysis procedure.

After some tests, the uranium recovery procedure adopted for its simplicity and acceptable efficiency is the following:

- the pellets loaded with the $UO_2F_2$ are placed in an Erlenmeyer, added 5 ml of distilled water, stirred during 3 minutes and then left to settle other 5 minutes;
- the solution so far obtained is separated and the extraction process is repeated 3 times more, the first one with distilled water and the two remaining with 1M $NO_3H$;
- with these 4 serial washing it is possible to recover around 85% of the original uranium, without dissolving an appreciable quantity of alumina. As the solution has certain turbidity due to the presence of very fine alumina in suspension, it is necessary to centrifuge the solution to separate the alumina.

This solution or an aliquot of it is evaporated to eliminate the fluoride that interferes in the isotopic analysis, and finally the dried product is dissolved in 1M $NO_3H$ to obtain a solution containing around 5 mg U/ml for the isotopic analysis.

### 3.5. Comparison of the isotopic analysis data

Available $UF_6$ samples were sub-sampled [3,4] in order to hydrolyze them directly for isotopic analysis. The isotopic analysis from the solution obtained by the UF6 direct hydrolysis method is compared to the isotopic analysis obtained from the alumina pellet method. UF6 standard samples were also utilized for this comparison between the two methods. Tables 3 and 4 present the results of this comparison.

No significant differences in the isotopic analysis by mass spectrometer were found between the direct hydrolysis method and the alumina pellet method. This is a logical result as the blank alumina pellet analysis by TXRF did not detect the presence of uranium. The presence of small quantities of alumina dissolved in the solution did not generate any difficulty in the isotopic analysis as well.

| Sample | U isotopic results actual method ($UF_6$ direct hydrolysis) (wt %) | U isotopic results new method ($UF_6$ alumina pellets sampling) (wt %) | Difference in the U-235 (wt%) |
|---|---|---|---|
| #1 | **U-235 = 1.468 ± 0.002**<br>U-234 = 0.0092 ± 0.0002<br>U-236 < 0.0007 | **U-235 = 1.467 ± 0.002**<br>U-234 = 0.0091 ± 0.0002<br>U-236 < 0.0007 | **-0.001** |
| #2 | **U-235 = 2.168 ± 0.003**<br>U-234 = 0.0146 ± 0.0003<br>U-236 < 0.0005 | **U-235 = 2.168 ± 0.002**<br>U-234 = 0.0145 ± 0.0003<br>U-236 < 0.0005 | **0** |
| #3 | **U-235 = 4.046 ± 0.005**<br>U-234 = 0.0378 ± 0.0002<br>U-236 < 0.0005 | **U-235 = 4.045 ± 0.004**<br>U-234 = 0.0378 ± 0.0002<br>U-236 < 0.0005 | **-0.001** |
| #4 | **U-235 = 1.751 ± 0.005**<br>U-234 = 0.011 ± 0.001<br>U-236 < 0.001 | **U-235 = 1.753 ± 0.003**<br>U-234 = 0.011 ± 0,001<br>U-236 < 0.001 | **0.002** |
| #5 | **U-235 = 1.624 ± 0.002**<br>U-234 = 0.011 ± 0.001<br>U-236 < 0.003 | **U-235 = 1.621 ± 0.002**<br>U-234 = 0,012 ± 0,001<br>U-236 < 0.003 | **-0.003** |
| #6 | **U-235 = 1.187 ± 0.002**<br>U-234 = 0.009 ± 0.001<br>U-236 < 0.003 | **U-235 = 1.183 ± 0.003**<br>U-234 = 0.009 ± 0,001<br>U-236 < 0.003 | **-0.004** |

**Table 3:** Isotopic analysis data comparison for $UF_6$ real samples

| Sample | U isotopic results actual method ($UF_6$ direct hydrolysis) (wt %) | U isotopic results new method ($UF_6$ alumina pellets sampling) (wt %) | Difference in the U-235 (wt%) |
|---|---|---|---|
| IRMM 020 | **U-235 = 0.210 ± 0.001**<br>U-234 < 0.003<br>U-236 = 0.029 ± 0.001 | **U-235 = 0.209 ± 0.001**<br>U-234 < 0.003<br>U-236 = 0.029 ± 0.001 | **-0.001** |
| IRMM 022 | **U-235 = 0.720 ± 0.002**<br>U-234 = 0.005 ± 0.001<br>U-236 < 0.003 | **U-235 = 0.720 ± 0.003**<br>U-234 = 0.005 ± 0.001<br>U-236 < 0.003 | **0** |
| IRMM 023 | **U-235 = 3.274 ± 0.004**<br>U-234 = 0.033 ± 0,001<br>U-236 < 0.003 | **U-235 = 3.268 ± 0.003**<br>U-234 = 0.033 ± 0,001<br>U-236 < 0.003 | **-0.006** |
| IRMM 029 | **U-235 = 4.173 ± 0.006**<br>U-234 = 0.080 ± 0.001<br>U-236 = 0.989 ± 0.001 | **U-235 = 4.165 ± 0.004**<br>U-234 = 0,079 ± 0,001<br>U-236 = 0.989 ± 0.002 | **-0.008** |

**Table 4:** Isotopic analysis data comparison for $UF_6$ standards

## 4. Future Work

As demonstrated in the previous section, the $UF_6$ sampling method using alumina pellets improves the safeguard measurement procedure applied to enrichment facilities.

Before implementing the new method as a routine procedure the following steps are foreseen as necessary:

- to perform a demonstration exercise at the enrichment facility in order to the operator evaluate its impact to the systems and operation;
- to certify the method by an independent international laboratory;
- to agree with IAEA, Operators and National Authorities for implementing the proposed $UF_6$ sampling method for U isotopic determination.

The first step is already being done at one of the enrichment facility laboratories in Brazil. The preliminary results indicate that the operator is comfortable with the method and is reproducing the positive results obtained previously at laboratory.

The second step will be performed by a laboratory out of South America that performs nuclear material measurements and is a nuclear material certifier.

The third step will be done after the conclusion of the two previous ones.

## 5. Conclusion

ABACC proposed a new method of sampling $UF_6$ for enrichment determination (ABACC-Cristallini Method) using a fluorothene tube containing alumina pellets that absorb and hydrolyze $UF_6$ directly during the sampling process.

The method was demonstrated at laboratory where an experimental system was set up.

The alumina pellets used are commercial catalyst support, and one gram of these pellets without any previous treatment was determined as enough for sampling $UF_6$ for enrichment determination.

The experimental results show that 10 to 30 minutes of contact of the UF6 with the pellets would retain enough quantities of uranium for isotopic analysis, even for system pressures lower than 10 mb.

The recovery of the uranium is simple and quick. It does not require any special equipment and it can be done in a radiochemical hood as well as in a laboratory bench, since no gas is liberated when the fluorothene tube is open.

The method shall be further qualified by the enrichment facility operator, and certified by an international independent laboratory.

The method responds appropriately and its implementation as routine procedure to enrichment facility safeguard does not present any foreseen difficulty.

It is evident the advantages that the new method presents compared to the actual method as: the $UF_6$ sample content left at the installation (archive sample) will be lower and less reactive, the laboratory procedures for manipulating the sample will be much easier, the residual uranium retained at the laboratory will be much lower, the sampling device is less expensive, there will be saves in transport cost as well, and it is relatively safer concerning radiological protection aspects during transportation.

## 6. References

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[2] Russell R. G.; *Application of Fixed Bed Trapping Technology for Removal of Low Concentration $UF_6$ from Plant Gaseous Effluent Streams*; Oak Ridge Model Conference, Proceedings; Volume I, Part 3, Waste Management; 1987, p.165-184.

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# Session 28

# NMA

# IMPLEMENTATION OF DOMESTIC REGULATION ON SMALL ONWERS OF NUCLEAR MATERIALS IN FRANCE EVALUATIONS AND PERSPECTIVES

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***Abstract:***

*In France, a detailed and clear regulation takes into account small owners of nuclear materials. The enforcement of the requirements stipulated by this system is ensured by the centralization of the data submitted by operators and by on-site inspections carried out by sworn and accredited inspectors under the competent authority. Even if the quantities are not significant, the listing of nuclear materials one can find is varied. And the small owners are not often well-versed of the regulation to apply. That's why the French national control on nuclear materials makes an effort to inform the small owners and sensitize them to the different constraints associated to their activities.*

*The first part of this paper will present the overall of the situation about the policy of the control by the French national authority. The second part will show the evaluation of the practice of control generated by the activity of a small owner, which are called "déclarant" in France. Finally the third part will present the new perspectives of the control by the French national authority.*

**Keywords: small owner = "déclarant"; declaration, inspection;**

## 1. First part: The overall of the situation

### 1.1 Definition of a small owner

Along with nuclear activities, small quantities are also used, outside the nuclear fuel cycle, in particular in industrial, medical and research sectors. Considering its nuclear situation and conscious of its national and international commitments in terms of national public security and nuclear non proliferation, the French government set up a national safeguard system under the authority of the Ministry in charge of Industry. It is the High Official for Defence and Security Service (Service du Haut Fonctionnaire de Défense et de Sécurité) which ensures protection and control of nuclear materials with the Institute for Radiation Protection and Nuclear Safety (Institut de Radioprotection et de Sûrete Nucléaire (IRSN)), acting as technical support body. This system is based on specific regulation which covers the entire civil nuclear field as well as the industrial, medical and research sectors. The basic aim of this regulation is to prevent or detect without delay the disappearance, loss, theft or diversion of nuclear materials, or equipment containing these materials regardless of their chemical or physical

form. The main text of this regulation, the Code of Defence, determines the current regulatory framework related to the protection and control of nuclear materials.
The decree n° 81-512 of 12 May 1981 specifies the different types of nuclear materials concerned with the French regulations, which are fissile and fertile materials and those identified as likely to be used in the confection of a nuclear weapon. They are listed as: plutonium, enriched uranium, natural and depleted uranium, thorium, tritium, deuterium, and lithium enriched in lithium 6. This decree also establishes three regulatory regimes, based on the nature and quantity of the nuclear materials involved, as presented in Table 1.
1) Licensing: for significant quantities of nuclear materials, a licence from the Ministry of Industry is required.
2) Declaration: below defined quantity thresholds of held nuclear materials, no preliminary licence is required but an annual declaration of undertaken activities.
3) Exemption: no specific requirements for the operator but the quantities of nuclear materials held must be very limited.

| | Plutonium, $^{233}U$ | Uranium ≥ 20% $^{235}U$ | Uranium < 20% $^{235}U$ | Natural and depleted uranium, thorium | Deuterium | Tritium | Lithium enriched in Lithium 6 |
|---|---|---|---|---|---|---|---|
| Licensing | >3 g | >15 g d'U5 | >250 g d'U5 | >500 kg | >200 kg | >2 g | >1 kg Li 6 |
| Declaration | ≤3 g<br>>1 g | ≤15 g<br>>1 g | ≤250 g d' U5<br>>1 g d'U5 | <500 kg<br>>1 kg | ≤200 kg<br>>1 kg | ≤2 g<br>>0,01 g | <1 kg Li6<br>>1 g Li6 |
| Exemption | ≤1 g | ≤1g | ≤1 g d'U5 | ≤1 kg | ≤1 kg | ≤0,01 g | ≤1 g Li6 |

Table 1: regimes of nuclear materials according to the decree of 12 May 1981

Finally, orders complete the legal framework. In particular, the order of 14 March 1984 stipulates technical arrangements related to the control, the accounting and the physical protection of nuclear materials under the declaration regime. It was modified on 21 May 2003 to specify the rules to be followed regarding supporting documents justifying inventory changes and inventory taking. A revision of these texts is scheduled in the next year.
It is worth noting that in France, safety and radiation protection matters are subject to specific regulations and authorities which include radioactive sources. These regulations also apply to most of nuclear materials, except Deuterium and Lithium that are not radioactive.

## 1.2 Activities of declarants

"Declarant" activities principally concern the medical, industrial and research fields. These activities are described in detail in table 2, for each type of nuclear materials. One can notice the diversity of uses of nuclear materials in non-nuclear fields.

| Type of nuclear materials | Uses | Fields |
|---|---|---|
| Depleted Uranium | Radiation shielding | • Industrial gamma radiography<br>• Radiotherapy |
| | Balance weights | • Aeronautics<br>• Oil-well drilling |
| Thorium | Welding electrodes | • Metallurgy |
| | Aircraft alloys | • Air museums |
| | Chemical Products | • Suppliers of chemical products<br>• Laboratories<br>• Pharmaceutics |

| Type of nuclear materials | Uses | Fields |
|---|---|---|
| Natural Uranium | Chemical Products | • Suppliers of chemical products<br>• Laboratories<br>• Pharmaceutics |
| | Dye for crystals | • Crystal manufactures |
| Deuterium | Solvents | • Laboratories, NMR (Nuclear Magnetic Resonance) techniques |
| Tritium | Radio luminescent devices | • Aeronautics<br>• Watch manufactures |
| | Radiotracer | • Hospitals<br>• Laboratories |
| Highly enriched uranium<br>Plutonium | Sources | • Source users |

Table 2 *:* main application fields of nuclear materials owned by "declarants"

As described below in section 1.3, annual declarations provide the authority precise figures on the inventory of nuclear materials held by the "small owners" population, which gathers French operators under the declaration and the exemption regime. Figure 1 provides an overview of this population regrouped by activity fields based on declarations of year 2005. At the end of year 2005, about 500 "déclarants" were identified in France, included about 195 operators holding nuclear materials under the exemption regime. These data are very similar for year 2006. Industrial radiography and medical radiotherapy users represent the highest population of operators placed under the declaration regime (more than 65%). On the contrary, the majority of operators working in the research field belong to the "exemption regime" and most of them are in possession of very small quantities of chemical products such as uranyl or thorium oxides, acetate or nitrates used for analyses, or deuterated solvents used in Nuclear Magnetic Resonance techniques.

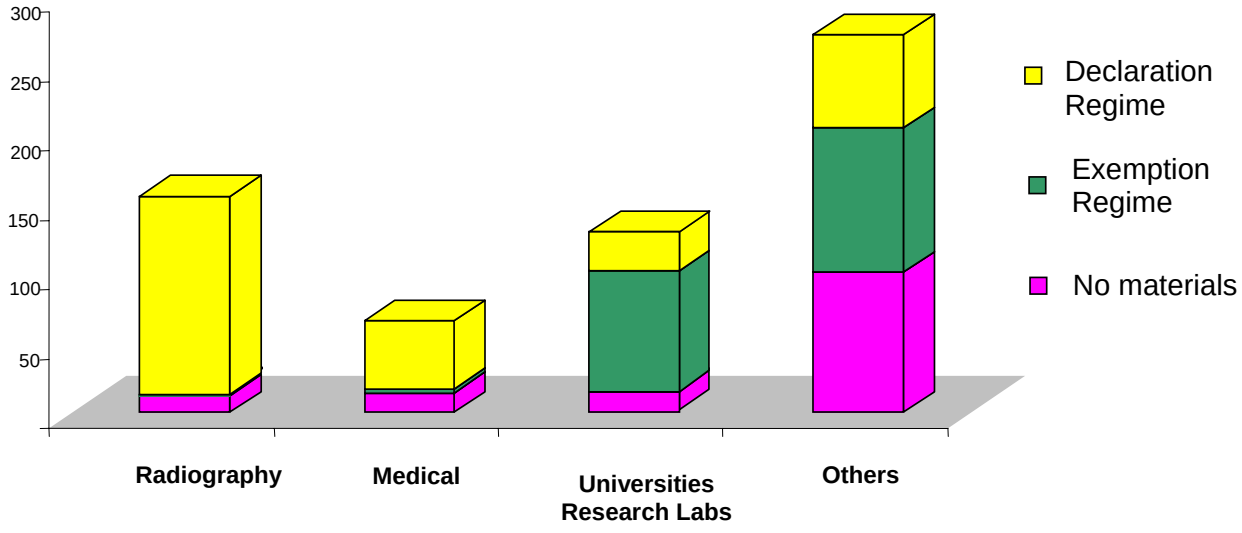


**Figure 1 :** status of operators subject to declarations regrouped by activity fields on 31 Dec. 2005. The category "No stocks" corresponds to operators who declared not to be in possession of any kind of nuclear materials.

As shown in figure 2, depleted uranium represents the main part of nuclear materials held by small owners. Others are much less significant. Most of the depleted uranium held by “déclarants” is actually encountered among industrial radiography users.

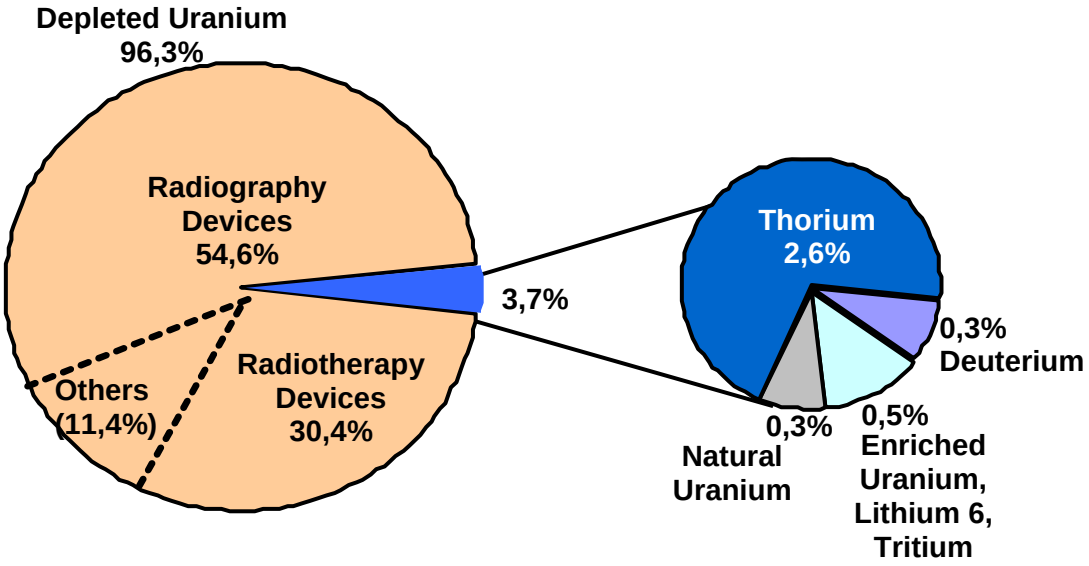


**Figure 2**: Mass distribution of nuclear materials held by declarants

### 1.3 Regulation requirements related to declaration regime

Prior to the receipt of nuclear materials under declaration regime, concerned operators must establish an initial declaration. This declaration, which is to be sent to the Institut de Radioprotection et de Sûrete Nucléaire , acting as technical support body for the competent authority, includes the following data:
· Identification data (company’s name, address, etc.) and the name of its owner and operator who are legally responsible.
· Type of activities concerned and localization of nuclear materials.
· Description of nuclear materials protection measures.

After this initial declaration, the small owner must set up a local accounting system and a paper copybook or a computerized system to keep records of all inventory changes. This step is essential for the traceability.
The operator must send to IRSN every year before January 31st, the following data :
- Inventory of nuclear materials as of December 31st of the previous year.
- Inventory changes occurred during the previous year, including the identification of shippers and recipients.
- Maximum inventory of nuclear materials and inventory changes expected for the present year.

Before filling the annual declaration, the operator must carry out a physical inventory to ensure that all nuclear materials present in the facility are correctly listed in accounting system. Inventory data must be provided for each category of nuclear materials defined in table 1.

Some small owners, however, does not send their annual declarations for their nuclear materials inventory and movements. Many different reasons have to be considered like company’s name or address change, person who is in charge of nuclear materials management change, sometimes the small owner “just” forget to send its annual declaration.
In the case of a decrease of the nuclear materials stock under the defined thresholds the small owner which become exempted owner, has to send a declaration.

That is the reason why the declaration process put in place by IRSN include at beginning of March, a first follow-up letter in order to recall to missing small owners that they have to meet the regulation obligations.

If the “déclarant” doesn’t answer to this first letter, a second one is send by the authority (HFDS) after the 15th May. One can consider that at the end of this step, most of small owners have sent their declaration to the national accounting group hold by IRSN.

### 1.4 Enforcement of declaration regime requirements

As previously mentioned, IRSN acts as the technical support body of the competent authority. Its missions include centralization of all declarations on nuclear materials, especially those submitted by owners of small quantities of nuclear materials. IRSN is also in charge of evaluating the declarations returned by “déclarants”. This evaluation consists of carrying out consistency checks with the previous declarations and making crosschecks and comparisons between the information transmitted by other “déclarants” or submitted by licensed companies.

On-site inspections are carried out by sworn and accredited inspectors under the authority of the Ministry in charge of Industry. They are also a pertinent tool in enforcing the declaration regime requirements. Technical visit programs are established after the analysis of annual declarations but also on the basis of specific events pertinent to a small owner or to a field of activities. These actions allow inspectors to carry out about 20 inspections a year at “déclarant” facilities.

The main points of an inspection are:

1 To remind the small owner of the national regulation related to the control and protection of nuclear materials.
2 To remind the small owner of the links between this regulation and others concerning radiation protection or radioactive management.
3 To check the compliance with the regulation and more particularly the order of 14 March 1984. In particular, inspections allow evaluating the local nuclear materials accounting system implemented by the small owners.
4 To check the documents related to these requirements.
5To check the correctness and the completeness of the previous physical inventory (with using portable detection devices adapted to the nature and the quantity of radioactive materials).
6 To analyze the arrangements made by the operator to ensure the physical protection of nuclear materials.

After the completion of an inspection, inspectors send a report to the competent authority. Based on those reports, the Authority send a follow-up letter requiring, if needed, corrective actions to be undertaken by the operators.

More than 300 on-site inspections carried out since 1995 have provided with a sound knowledge of the use of nuclear materials held by small owners in France. This allows IRSN to play a main role in the preparation and the implementation of the regulatory documents concerning the use of nuclear materials in the medical, industrial or research sectors on behalf of the Authority. It also allows IRSN to identify specific issues and raise the attention of the Authority whenever is necessary.

## 2 Evolution of the system and evaluation of the control on small owners

The organisation set up to follow the “déclarants”’ activity is globally effective and it allows a follow up of the small nuclear materials owners which are for the most of them out of the nuclear fuel cycle. They do n’ot have the same security culture than the nuclear operators and nuclear materials are not there key activity but have some devices used for other matter.

IRSN and HFDS wondered about the reasons that some small owners do not send, or with difficulty, their annual declaration.
A new criterion put in place in close link with the nuclear accounting group allow the early detection of small owners which are not sending their declaration form in the time delay prescribed by the regulation with no regards to follow-up letters. Those small owners have priority in the inspection program. A half dozen of those small owner’s inspections were carried out under this new criterion and were very fruitful.
So, one of these inspections allowed the authority to discover some undeclared nuclear materials (thorium and enriched uranium used for testing source). The small owner owned these nuclear

materials since 1975, 6 years before the implementation of the nuclear materials' regulation. The declaration routine was lost when the person in charge of nuclear materials management retired.
Besides, a scrap merchant obtained an aircraft wreckage among other metal pieces. By the use of a gamma ray detector during the inspection a reactor containing pieces in alloy of thorium was identified. The previous aircraft's owner did not warn the scrap merchant about the presence of this alloy of thorium, so this nuclear material was not recorded in the national accounting system.
In another example, is a firm in liquidation whose activity was the breaking up of gammagraphy device. This firm kept these devices expecting from an appropriate storage centre. Now, this source is being about to be removed to be safely and security stored.

These examples show the interest of the inspections in small owners, who don't size up the risk inherent to nuclear materials possession. Whatever it could be the follow-up of the physical inventory, the accounting system, physical protection or the radioprotection.

During our inspections we met a lot of small owners which ask some help to give them a solution to evacuate their pieces in depleted uranium.
A useful note is that since the 1st January, a "working group" gathering Security and Safety Authorities, the National Radioactive Waste Management Agency, and the CEA which is a French government-funded technological research organisation, was constituted and put in place an official channel allowing small owners to be relieved from their depleted uranium and from the regulatory system too.

## 3. Perspectives

The evolutions shown in the last chapter is the first step of the process witch aims to enhance the effectiveness of the control and to optimized human resources which wouldn't increase in the future.
In France, three persons are in charge of the control of small owner of nuclear material at IRSN and one in the Ministry of industry, they take care of about 300 small owners holding nuclear materials as described before, they rely on the French national accounting group to receive, process the annual declaration, support small owners during the declaration process and to gather needed data to prepare their action.
The French regulation should evolve in the next months and this should have some consequences on the actual small owner regulation. The small owners should ask to the national authority for the authorization of owning nuclear materials. The authority will have a period of time to assess the request and, if needed, to ask for complementary information. A formal authorization will have to be given prior to any reception involving nuclear material.
Moreover it was decided to improve the process of inspections like it was shown in the chapter 1.4.
We have decided to separate in two parts this process. The first one, the inspection,, we can focus on the checking of the compliance with the regulation and the real situation. In other words, we will be concentrated on the evaluation of the nuclear materials accounting, the physical protection, and the physical inventory of nuclear materials.
All of is concerned by the regulation explanation (cf. point 1 and 2 from § 1.4) will be treated in another part .,like symposiums. In fact, although the text of the order of March 14$^{th}$ 1984 is send by the national accounting group to all small owners with the annual declaration form. In 80% of the inspections, the small owner has not got a good knowledge of the regulation. The explanation is that either they do not take the time to read it or they have some difficulties to understand it.
Considering this experience feedback, the Authority decided to organize each year, from 2008, two or three annual meetings of information regrouping small owners of a region. The goal is to cover most of the country in three years. In addition to a presentation of the regulation, the accounting system and contents of unfolding would be detailed. In addition, those meetings could be useful to answer small owners questions and by having a better knowledge of there concern to enhance the declaration and control process.
When the country will have been covered, taking on to account the renewal rate of small owners, the frequency of those meetings could be reviewed.

## 4. Conclusion

In France, a detailed and clear regulatory system has been set up for small owners of nuclear materials. The enforcement of the requirements is ensured by the centralization of the annual declaration data submitted by operators and by on-site inspections carried out by sworn and accredited inspectors under the authority of the Ministry of Industry.

The small owners which are around 500 and for most of them outside the fuel nuclear cycle do not have much concern with nuclear materials. Specific attention has however to be put on this nuclear materials which could be subject to theft, diversion or malevolent actions. That is the reason why France, since a long time, has decided to ensure the follow up of these nuclear materials by the declaration regulation.
But according to our experience, the Authority decided to enhance the control approach by adding criteria in inspection targets selection, enhance the regulation objectives. Requirements understanding will be ensured by the organization of dedicated meetings, symposiums, all over the country. By those means, the Authority wished to contact most of the “déclarant” to explain them the requirements of the regulation. Soduring the on-site inspections, the inspectors can focus their control on the nuclear materials management organisation, and their physical protection. Setting up such a practice of the control on small owners will promote the security culture in facilities which are outside the nuclear fuel cycle.

# Small Users – Big Headache

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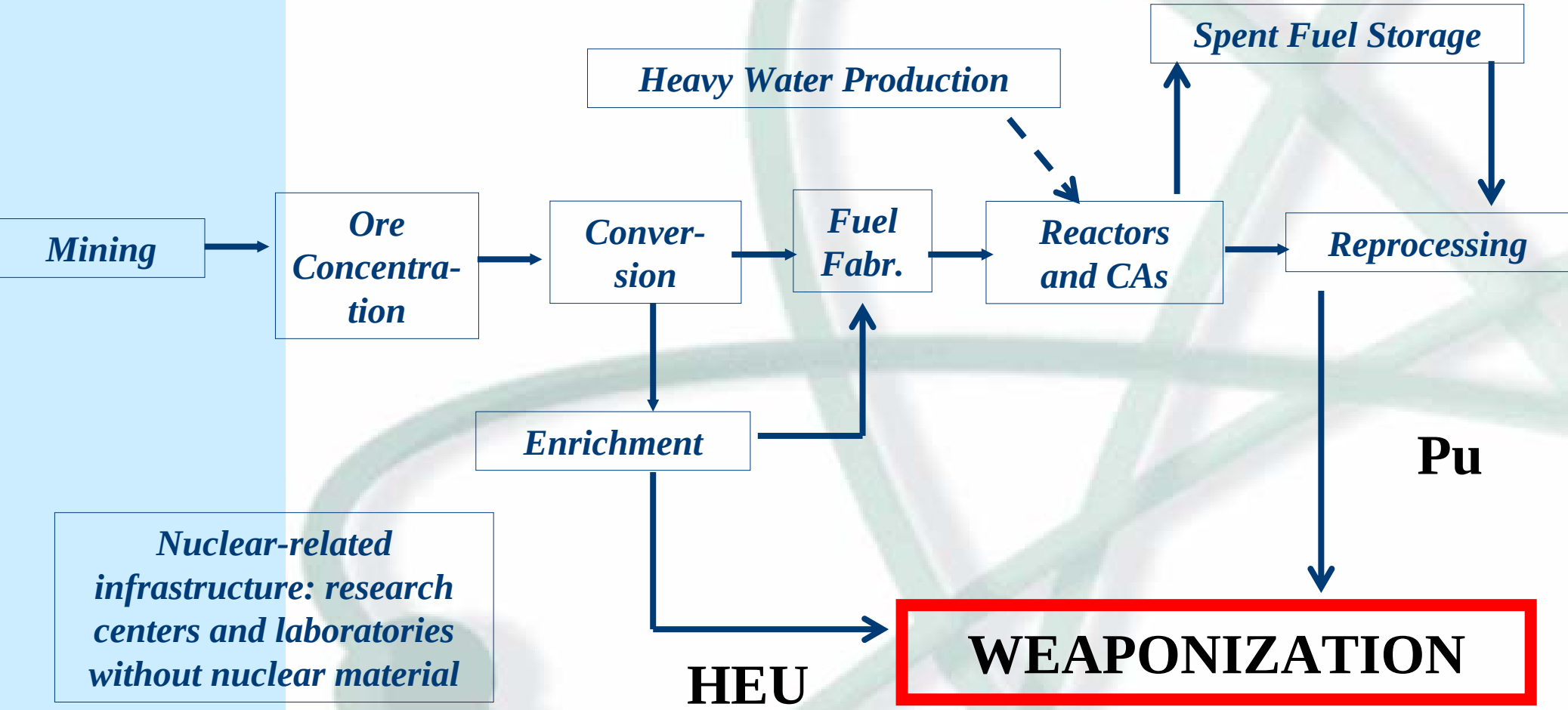
29th ESARDA Annual Meeting

Aix en Provence, France

May 22-24, 2007

# Nuclear Fuel Cycle



# Non-Nuclear Applications

## Nuclear Material in:

- Medicine
- Industry
- Education
- Other use

# NPT

- Article III.1
  - Safeguards applied with a view to preventing diversion of nuclear energy from peaceful uses to nuclear weapons
  - The safeguards required by this Article shall be applied on all source or special fissionable material in all peaceful nuclear activities

# Safeguards Agreements

- Exemption from Safeguards based on use
  - Special fissionable material, when it is used in gram quantities or less as a sensing component in instruments
  - Nuclear material, when it is used in non-nuclear activities, such as production of alloys and ceramics, if such nuclear material is recoverable
  - Plutonium with an isotopic concentration of plutonium-238 exceeding 80%

# Additional Protocol

- Provision of information on exempted material (Article 2.a.(vii)(b) )
  - Nuclear material, when it is used in non-nuclear activities, but not yet in a non-nuclear end-use form

# Euratom Treaty

- Article 77
  - The Commission must ensure that, in the territories of the Member States ores, source materials and special fissile materials are not diverted from the intended uses as declared by the users.
- Article 84
  - In the application of the safeguards, no discrimination shall be made on grounds of the use for which ores, source materials and special fissile materials are intended.

# Regulation on the Application of Euratom Safeguards

- Article 1 Scope
  - It shall not apply to holders of end products used for non-nuclear purposes which incorporate nuclear materials that are in practice irrecoverable

# MBAs in Lithuania

# Small amounts of Nuclear Material in Lithuania

Plutonium or uranium present in:

- Smoke detectors
- Other devices (neutralizers, level gauges, calibration courses, etc.)
- Shielding devices

# Associated Problems

- Determining nuclear material weight and isotopic composition
- Establishment of accurate inventory, especially of disposed items
- Provision of access for the inspectors to see, count and make measurements

# Conclusions

- Small amounts of nuclear material in non-nuclear use hardly pose proliferation risk
- National and international control of small users adds to strengthening the nuclear security

# UK/Russian Collaboration on NMAC Systems at Gas Centrifuge Enrichment Plants

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29th ESARDA Annual Meeting
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***Abstract:***

*As a result of co-operation between Angarsk Electrolysis Chemical Complex (AECC) and Urenco (Capenhurst) Limited, an up-to-date computerised Nuclear Material Accountancy and Control (NMAC) system has been created at the AECC enrichment plant. Representatives of the UK government, the Department of Trade and Industry (DTI), and the agents authorised by DTI – Urenco (Capenhurst) Limited and British Nuclear Group – provided technical assistance for the development of the system, and supplied the required equipment and software, to the value of £308k. Experience gained during the years of cooperation in the field of NMAC was extensively used by AECC to build this computerised system, which meets the requirements of the Federal Information System of the Russian Federal Atomic Energy Agency, and could readily be modified to meet IAEA requirements.*

**Keywords:** Russia, UK, NMAC, centrifuge, enrichment

## 1. Introduction

Over 90% of gas centrifuge enrichment plant capacity in the world is based in the Russian Federation (owned by the Federal Agency for Atomic Energy) and in Germany, Netherlands and United Kingdom (owned by Urenco Enrichment Company Ltd). There are many similarities between the two companies' plants: they are large (each being over 1000 tSWpa capacity); they are at many sites (Angarsk, Novouralsk, Seversk, Zelenogorsk, Almelo, Capenhurst and Gronau): they use centrifuges for separation of uranium isotopes (with very low pressures and hence a low hold-up of uranium in the cascades); they separate natural uranium hexafluoride (UF6) into a product enriched in U235 (for manufacture into fuel for nuclear power stations world-wide) and a tails material depleted in U235 (kept as a strategic reserve for potential re-enrichment) . Moreover, the governments of the four countries are equally interested in having a tight control on the use of nuclear material and technology, and thereby promoting non-proliferation aims.

There are however some differences: the Urenco plants have been subject to international safeguards verification by Euratom and IAEA for over 20 years, whilst the Russian plants are not – instead they are controlled by Moscow; and although the Russian plants are larger than Urenco's, they have had less investment in modern computerised systems. It was not surprising that when asked by the Russian government in 1997 for assistance in modernising the nuclear materials accountancy and control (NMAC) system at the enrichment plant at Angarsk, the UK government (backed by technical support from Urenco) readily agreed. The aim of the Russian government was to have an NMAC system at Angarsk capable of meeting IAEA standards, should it ever be decided to offer the plant to IAEA for safeguards verification. There would then clearly be scope for replicating the system at the other Russian plants.

## 2. Chronology of Project

### 2.1. Laying the Foundations

Between March 1996 and May 1997, there took place a joint UK/Russian/Chinese collaborative project, at which Urenco and DTI gave advice to assist the implementation of an IAEA safeguards regime at the Russian-designed gas centrifuge enrichment plant then under construction in China – IAEA and BNFL also assisted in this project. The success of this collaboration led to a request by the Russian Ministry for Atomic Energy (Minatom) for assistance in improving the NMAC system at the gas centrifuge enrichment plant at Angarsk, in eastern Siberia. The basic principles were agreed at an intergovernmental meeting held in Moscow in April 1997, at which representatives from AECC and Urenco also attended. Recognising that neither the British nor Russians are great linguists, steps were taken to improve communication: interpreters were appointed, some staff took language courses, and a British/Russian technical glossary of terms relating to the application of NMAC in centrifuge enrichment plants was written. Finally, in July 1998, DTI secured funding from the UK government, to allow the project to proceed.

### 2.2. Gaining Understanding of NMAC Systems

Between November 1998 and June 2001, a series of seminars took place at Capenhurst and Angarsk, at which there were thorough demonstrations of the NMAC systems at both sites, and a detailed report was written, which described the systems in some detail. A set of criteria was jointly developed for assessing NMAC systems – see section 3 of this paper. A fully working copy of the Urenco NMAC system was shipped to Angarsk, on a Sun Unix computer, running an application developed using the PI/Open database package. This was modified by AECC to provide a prototype NMAC model of the Angarsk plant – from this, the AECC staff could fully understand the concepts employed by Urenco.

### 2.3. Developing the NMAC System at Angarsk

In 2002 and 2003, AECC prepared a full technical proposal for the new NMAC system to be developed. This defined the scope and functionality required of the system, and the cost, effort and timescale needed to develop the system. During 2004 and 2005, a detailed design of the proposed system was prepared by AECC, with a peer review being carried out by Urenco. In 2005, a detailed listing of hardware and software to be provided by DTI was agreed between AECC and Urenco. DTI appointed British Nuclear Group (Project Services) Ltd to obtain this IT equipment; after a tendering process, a Russian supplier was chosen, and all the equipment was delivered to Angarsk in March 2006. During 2006, AECC installed the equipment on site, developed the required application software and brought the computerised system into operational readiness.

In January 2007, meetings took place at AECC attended by representatives of DTI, Urenco (Capenhurst) Limited and BNG Project Services. AECC demonstrated the functionality of the system using test data to a test programme developed jointly by AECC and Urenco. The demonstration was witnessed and verified by Urenco, who concluded that:

- The equipment and software delivered to AECC has been brought into operation. This is a well designed system incorporating high availability and secure features.
- The system developed meets the requirements for nuclear materials accountancy and control at the enrichment plant.
- The reports generated by the system meet the requirements of the Russian state NMAC system. In order to meet IAEA requirements, additional reports would be necessary. These could easily be generated from the data captured by the system.
- The system is ready for operation.

AECC intend to bring the system into full operational use in all material balance areas (MBAs) on site, during the course of 2007.

## 3. Agreed Criteria for Assessing NMAC Systems

In 1999, Urenco and AECC jointly developed the following criteria, for assessing the performance of NMAC systems. In particular, it was agreed that the new system to be developed at Angarsk would be designed to take account of these criteria:

### 3.1. Degree of Compliance of the NMAC System with Requirements of National Regulations and Company Quality Systems

Number of anomalies, not foreseen by technical documents, appearing between physical inventory takings (PITs).

### 3.2. Quantity, Education Level and Experience of NMAC Personnel

Number and educational level of NMAC personnel. Number of NM operators with NM work experience between 5 and 10 years (as a percentage of the total NM operators).

### 3.3 Security of NMAC System with respect to Accidental Loss or Distortion of Data

Subdivisions, services and their functions of carrying out random inspections of the NMAC System in MBAs in parallel with the plant special control regular groups.

### 3.4 Security of NMAC System with respect to Unauthorised Activities

Factors preventing unapproved removal of NM from an MBA during the calendar year.

### 3.5 Data Collection

Speed of transmission of source data to the accountancy department: average time taken from generation of source data to entry of that data into accounts. Number of occasions the data relating to movements of NM is manually transcribed from generation of the source data to entry into the accounts. Number of errors in the data relating to movements of NM incorrectly recorded and their frequency (as a percentage of the total number of container moves performed with undetected errors in data). Number of errors in data relating to movements of NM not recorded and their frequency (as a percentage of the total number of container moves not being reported to the accountancy group or department).

### 3.6 Preparation of Accounts

Timeliness of accounts: period of time required to obtain the account balance on the closure of the accounts. Restriction on movements of NM down to a minimum at the time of physical inventory taking.

### 3.7 Structure of the NMAC Systems

Completeness of all site NM registration in accounts showing receipts/issues/balances as opposed to being held on a stock list. Presence of technical documents determining the precision of NM inventory taking in an MBA. Perfection of the methods for measuring acceptable material misbalance. Degree of data receipt automation for between-the-balances period. Obligation to maintain "double entry" accounts for both issue and receipt MBAs (for U and $U_{235}$). Maintenance of accounts in the NM process areas for material unaccounted for (MUF) and discards. Maintenance of accounts in NM storage areas for shipper/receiver difference.

### 3.8 Preparation for Physical Inventory Taking

Plenary powers of inventory committee. Definition of the application competence of technical documents, determination of inventory procedures, definition of the maximum acceptable calculated misbalance at the NM inventory taking in an MBA.

### 3.9 Physical Inventory Taking

Frequency of taking physical inventories in both process and storage areas. Proportion of estimated material in the process area inventory. Simultaneity of physical inventory taking in all production MBAs. Proportion of estimated NM in the MBA not subject to measurement during physical inventory taking. Possibility of generating stock lists corresponding to account balances. Speed of carrying out an inventory in both process and storage areas.

### 3.10 Evaluation of MUF

Presence of action levels set for material unaccounted for (MUF) assessment in the process areas. Identification of the degree to which systematic and random errors affect the misbalance in each balance period. Size of action level as a percentage of plant inventory or throughput. Size of MUF as a percentage of plant inventory or throughput.

### 3.11 Generation of External Reports

Efficiency of creating external reports. Data stated in external reports. Decision taking regarding external reports.

## 4. Description of new NMAC system at Angarsk

### 4.1 Local Area Network

The Local Area Network (LAN) encompasses three main areas at AECC: the enrichment plant, storage areas, and offices where reports are generated. The cable infrastructure of the computerised NMAC system was built using fiber-optic cables. The total length of LAN cables is 9,810 metres. All 22 client workstations are interconnected by this network and are connected to a single server centre that stores the informational part of the system. Standard PCs without hard drives are used as terminals for the computerised NMAC system. The operating system and application software are loaded from the servers installed in the server centre.

### 4.2 Server Centre

The server centre of the computerised NMAC system is located in an isolated, specially equipped room housing servers, passive and active network equipment, as well as power supplies, air conditioning, a fire alarm system and protection from unauthorised access.

The server centre houses:

- Five servers:
  - Database server (ORACLE 9.2i is used as RDBMS)
  - Application server (BAAN IV ERP system is used as the main application software)
  - Terminal server, additionally acting as the primary domain controller
  - Backup server, additionally acting as a spare domain controller and as a host for LAN management system (Optivity LAN Management software is installed on that server)
  - Training room server, additionally acting as a terminal server, database server, application server and domain controller for the training room terminals
- Data storage
- Tape library
- Main router of the computerised NMAC system LAN
- Router of the computerised NMAC system training room LAN.

### 4.3 Design of NMAC System, and Security Features

Just like any similar system, this system provides for accounting of all operations relating to nuclear material. This accounting is based on accounting records. Once an operation is

complete, the personnel must record their activities in the system. For this purpose, they enter into the system the operation carried out and those characteristics which have changed as a result. Such characteristics include:

- Location (when a movement has been performed)
- Material weight (when weighing has been performed)
- Isotopic assay (when mass spectrometry analysis has been conducted)

Since the creation of an accounting record is the crucial aspect in accumulating NM-related information, arrangements have been made to ensure that data entered is properly verified and approved. As a rule, three people participate in the creation of an accounting record:

- The first person enters the data
- The second person verifies the data
- The third person approves the data.

When an accounting record is made, the system exercises control over the data entry. The user who enters a record can only enter those operations which are pre-defined in his or her job description. The system is configured so that a specific worker can only perform specific operations. In addition, it is only possible for a user to enter information which relates to a container located at the corresponding stage of the process, and it is only possible to move a container to those locations which are accessible from the given workplace.

Verification and approval of the records entered can be done only by personnel authorised to perform such operations. Only when an accounting record has been approved can it be used in subsequent activities, such as generation of reports and datasheets.

The mechanism of accounting record creation described above makes it impossible to omit an operation. For example, the absence of a container movement operation would prevent the next participant from selecting the corresponding container to create an accounting record (since that container would not be found in the list of actual storage places). This way, the system automatically controls the personnel activities. This significantly improves the reliability of NM accountancy.

The next important aspect of the system is the personalisation of use of the system. Each user logs on to the system using a unique ID which must be confirmed by a password. After this authentication, a menu is shown on the computer screen which contains only those operations accessible in the system for that user.

Internal auditing is routinely carried out by the system, i.e. there is an audit log which automatically records all activities performed by users.

The main menu is not the only level where user rights are restricted. The system also provides for restriction of rights at the level of specific parameters. Some workers are allowed to enter data; other workers can only view data. Also, access can be restricted to specific types of records in the database. The way that access is granted or denied is defined by the system administrator, depending on job descriptions, orders, etc.

## 4.4 Training

Taking into account that personnel work in three shifts, 24 hours a day, there are about 80 people using the NMAC system. A training room has been provided as a part of the system. The training room is a small room with three PCs connected to a separate database. Any user involved with NMAC can take both basic and advanced lessons in that training room in accordance with the duties that are defined for that person. An administrator PC is located in the training room. The administrator can always give advice to those taking lessons.

## 4.5 Application Software

The application software of the system was developed by AECC specialists. They faced a challenging task: to develop the system rapidly - within one year. A decision was made to build the system using a BAAN environment because it had already been used for building

systems for the management of the company. The BAAN system comprises many modules covering conventional areas of activities of any company. But since nuclear material accountancy is a specialist field, there is no corresponding module in BAAN. AECC specialists developed a module which allows the solution of all basic problems relating to NMAC at the company.

## 4.6 Reports Generated

The system developed makes it possible to generate reports on the availability and movements of NM for any period of time specified by the user. All generated documents can be categorised as either accountancy reports or operating documentation.

Accountancy reports are documents which are mandatory for submission by the company to external organisations. These are: stock list, inventory change report (ICR), inventory taking report and material balance report (MBR). These have all been designed to comply with the requirements of the Federal Information System of the Russian Federal Atomic Energy Agency. The reports do not quite meet IAEA reporting requirements. However, if operations at AECC in the future were subject to IAEA safeguards, it would be straightforward to create suitable additional reports, as the extra data needed is already being captured by the system.

A user can also obtain operating reports from the system, such as: advice note, NM container movements history, operations history and nuclear material certificate. In addition, custom reports that can be built by a user based on the data accumulated in the system.

Examples are of various reports are given in Figures 1 to 3; these use dummy data, as data on actual holdings of nuclear material are not permitted to be published.

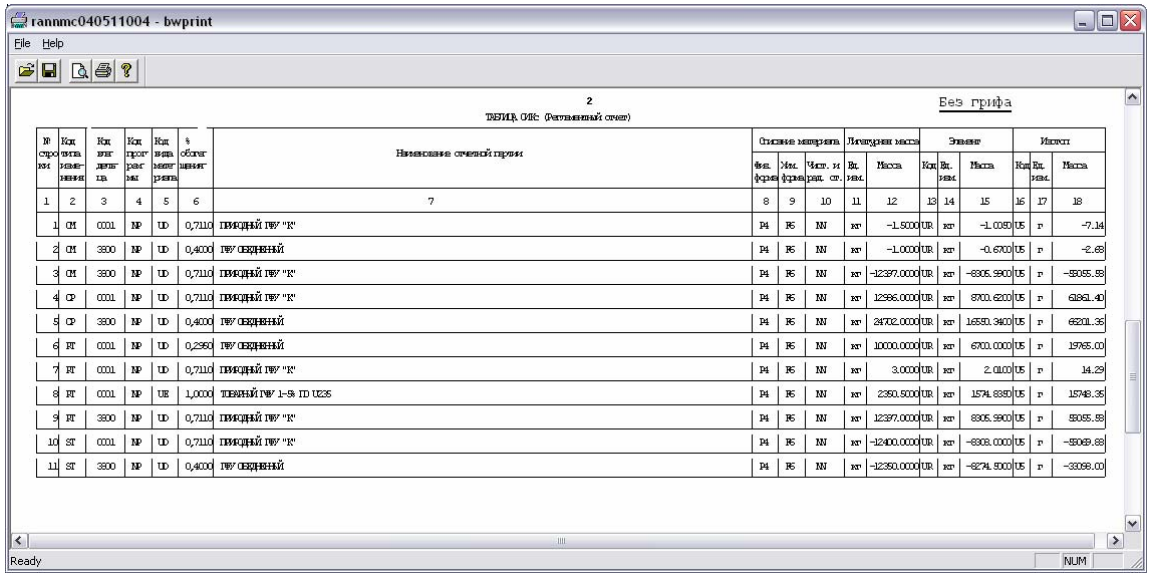


**Figure 1:** Inventory Change Report

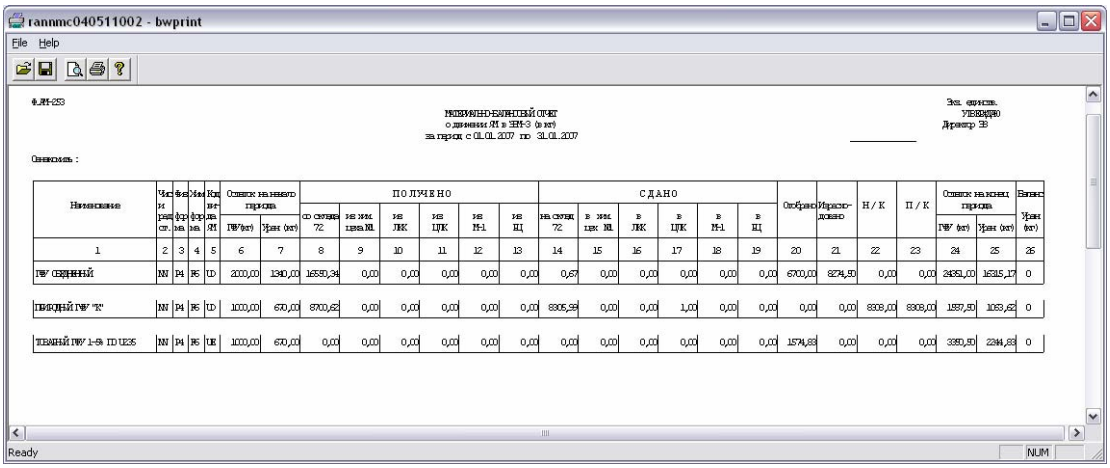


**Figure 2:** Material Balance Report

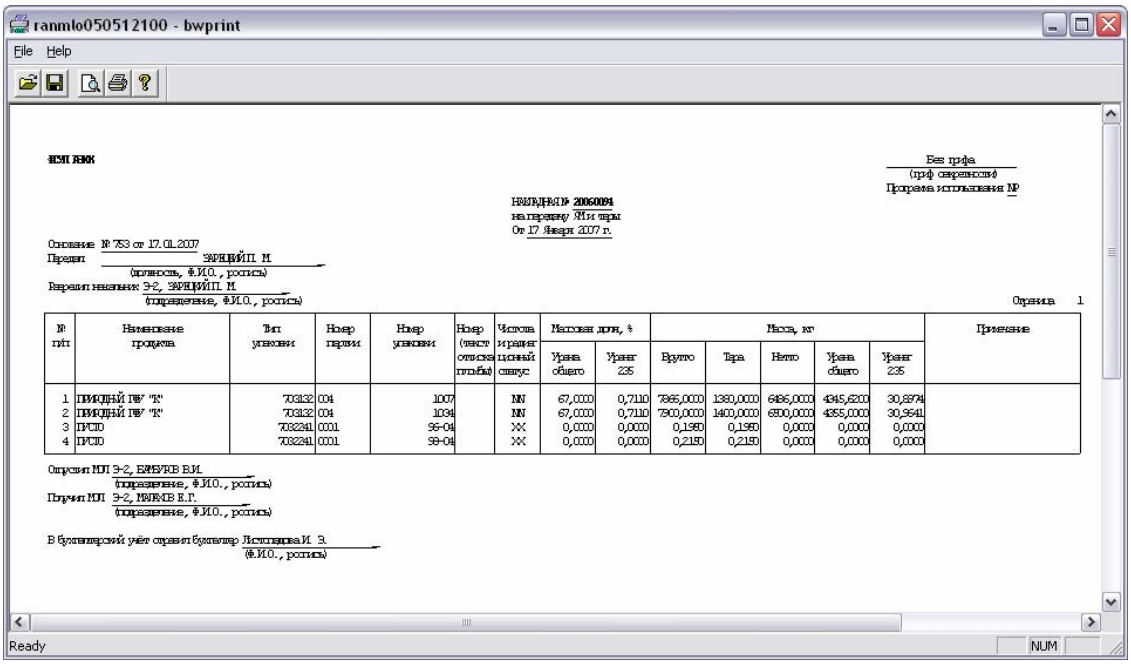


**Figure 3:** Advice Note

## 5. Acknowledgements

The UK Department of Trade and Industry (DTI) funded the project: for effort provided by the UK parties, travel and accommodation costs, and for supply of computer equipment and software.

British Nuclear Group (Project Services) Ltd procured the computer equipment and software.

Both UK DTI and the Russian Federal Agency of Atomic Energy provided political support throughout the project.

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# Session 29

# Measurement data and modelling

# A Perspective on Undeclared Inventory Measurements

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***Abstract:***

*Quantification of holdup or undeclared inventory, the residual nuclear material remaining in process equipment, has long been a challenge to those who work with nuclear material accounting systems. Fortunately, nuclear material has spontaneous radiation emissions that can be measured. Unfortunately, it can be quite difficult to relate this measured signal to an estimate of the mass of the nuclear deposit. Simplified analysis models, unknown distribution of the nuclear material, uncertainty about the intervening attenuators, background(s), and the source-to-detector distance(s) can have significant impacts on the quantitative result.*

*The quality of the quantitative assay result is intrinsically part of the answer and must be understood and used by any system or person making use of the quantitative results. Measurement quality is often divided between two descriptors, precision, and bias. Specifically, the uncertainty in the measurement results can be divided into these two categories. Precision is used to describe the reproducibility or random error component, while bias is used to describe the systematic error component or the concept of "average difference from truth." Unlike other nondestructive assay measurements, counting statistics is typically not the primary source of measurement error, consequently propagation of error calculations based on counting statistics generally underestimates the uncertainty in holdup measurement results.*

*This compilation of measurement experience from the last three decades for Special Nuclear Material (SNM) processing facilities documents the possible quality of the measurements of SNM process holdup.*

**Keywords:** Perspective, undeclared inventory, holdup measurements

## 1. Introduction

Holdup measurements are challenging, in part, because they are not performed in a controlled environment such as a laboratory. They do not present standardized containers, the measurement geometry is uncontrolled, and they are generally performed in the nuclear material process areas surrounded by unknown and uncontrolled backgrounds. Holdup measurements are often considered an art. Alternatively, they can be considered an underdetermined mathematical problem. Either way, these characteristics indicate that it may be difficult to reliably estimate holdup measurement uncertainties.

One should first consider the definition of holdup. It has been defined as undeclared inventory, in-process inventory, a residue remaining after cleanout, or the process contents before cleanout and recovery. While this presentation is pretty much independent of which definition is used, specifically it addresses the other factors in estimating holdup measurement uncertainty, clearly much confusion or error can result from a discussion between two parties with different definitions of the nuclear material holdup.

Most mature measurement methods in regular usage in the nuclear fuel cycle undergo evaluations to assess measurement errors.[1] Sufficient data now exists to include an assessment of holdup errors. This paper documents a variety of holdup measurement experience in facilities for HEU, Pu, and LEU. In addition, experience from training courses using calibration materials to simulate holdup is included.

## 2. What is a Holdup Measurement?

Before one can assess measurement uncertainty, one must define the measurement. In addition to the assorted definitions of holdup, holdup measurements have been described in various ways including:

- a treasure hunt,
- detective work,
- on-the-spot improvisation, or
- hundreds of detailed, repetitive (and possibly boring) measurements.

These descriptors indicate the expectations one might have regarding the uncertainty in a given situation. In the authors' experience, holdup measurements are never requested just to spend money or keep personnel busy. They are requested in the hope of saving money or of finding a valuable asset. The resultant measurement uncertainty is coupled with how much the customer wants to pay. A rapid scan for hot spots will likely have larger uncertainty than methodical, replicate measurements that cost more to perform. However, there is no guarantee that increased funding and effort will provide an improved rate of return. The fuel cycle has ample examples of increased funding not yielding the desired improvements. With the additional measurement challenges found in holdup measurements, the use of highly skilled professionals is one way to improve the odds of getting better quality results, with reduced measurement uncertainties.[2]

## 3. Why Measure Holdup?

The desired uncertainty can be related to the amount of money to be spent performing the measurement, while the amount of money available is related to why the measurement is being performed. Possible reasons for performing holdup measurements include:

- Economics
- Criticality safety
- Health hazard (D&D), and
- Safeguards

Special nuclear material is valuable. Some estimates place its cost per gram higher than gold or platinum. Facilities find it important to be interested in economics, in where their valuable assets are, and in how easy it is to retrieve them for use. If the cost to retrieve and use the asset is lower than the cost to buy new assets, and there is sufficient funding to retrieve the material, then the facility is generally interested in retrieval. The uncertainty in how much there is to retrieve is a component in this decision.

If the reason for measuring holdup is criticality safety, one must improve the measurement uncertainty as more material is located, particularly if kilogram quantities are localized in unsafe geometries. However, if the quantities being measured are orders of magnitude below the criticality safety limits, large measurement uncertainties, with correspondingly lower measurement costs, are acceptable.

Many nuclear material accounting systems assume that measurements have a constant percentage error, and the holdup results are expected to fit into this same simplistic model.

Overlaid on these considerations is the limitation of budget. What is the customer able to pay and what is he willing to pay for. Before starting measurements, it is important to clarify the customer's needs and expectations with respect to the assay uncertainty.

Many sources of measurement error are present independent of the expectations or future use the customer might have for the measurements. However, the customer's expectations (and budget) can play a significant role in how much effort is expended in considering sources of error.

## 4. Sources of error

The most significant source of error in nuclear holdup assay is usually the lack of knowledge about the geometry of the deposit being measured. Incorrect guesses about the material location and

distribution, followed by guesses about the intervening attenuators can lead to significant bias in the reported results. While the bias can be either positive or negative, experience usually shows that the reported results are biased low.

There are several models for acquiring and analyzing holdup data;[3-5] all have simplifying assumptions that may cause significant bias. A lack of well characterized or representative calibration materials can be treated adequately. Incorrect treatment of background is often not recognized until after the results are reported. Counting statistics can be handled using standard techniques, consequently it is usually the smallest source of uncertainty and occasionally the only one reported.

When corrections are made for these or other effects, it is useful to keep track of the possible errors due to incorrect assumptions. In general, it is prudent to remember that one has less control over the measurement process and the measured item than one has in a laboratory.

## 5. Means To Estimate Measurement Uncertainty

Assuming one has the resources and funding, there are several means to determine holdup assay uncertainty. Some can be applied during the measurement, some are based on (facility-specific) experience, and the best relies on actual cleanout and recovery of the nuclear material during the assay campaign. Then the cleaned-out items are remeasured and the mass difference compared with the cleanout mass. It can be hazardous to apply bias corrections to holdup measurement results based on cleanout results. The better approach is to improve the measurement procedure and the analysis model.

Replicate measurements can give information about precision. Sometimes adding more measurement points per item can yield information about the suitability of the analysis model. Alternatively, if one measures the item from several directions or several source-to-detector distances, suitability of model specific parameters or geometries can sometimes be assessed.

Intelligent guessing and the experience of subject matter experts can be useful in estimating measurement uncertainty. Many operators have experience showing where significant deposits have previously existed, and many measurement experts can apply lessons learned from other situations. The input from subject matter experts who have been able to adjust their measurement results and procedures based on cleanout values is invaluable.

## 6. Reported Accuracy

The precision or random error can be readily determined for all NDA measurements including holdup. Because of the many measurements performed, the overall precision of holdup measurements is usually of the order of a few percent or less. However, the accuracy or systematic error is very difficult to determine, because it is difficult to know the true mass of nuclear material held up in the equipment of a complex facility. Often, the accuracy estimate for a holdup campaign is simply the "best guess" based on judgment and experience. Such estimates are typically in the range 25 - 50%, because of the many unknown factors and assumptions required to calculate the nuclear material mass. In some cases, e.g. glove boxes, known standards can be introduced and measured in addition to the holdup. In a few cases, an effort was made to clean out and recover the measured material that was then analyzed destructively and compared with the measured holdup. A complete clean out is usually difficult and costly, but this is the best way to determine holdup assay accuracy.

In the early 1980s, a holdup measurement campaign was conducted at a shut down part of the Portsmouth Gaseous Diffusion Plant (PGDP) in Piketon, Ohio. Gamma-ray measurements were made with a collimated NaI detector and neutron measurements with a portable slab detector. A total of approximately 250 stages (converter, cooler, compressor, and piping) were measured during the campaign. Afterwards, three cells (12 stages each) were cleaned out and the uranium recovered. The U was also measured and recovered from an isolated converter. The results from this are summarized in Table 1. Because the gamma measurements only covered the converters, they should only be compared with the neutron assay of the isolated converter. These results are typical of what one finds in such holdup studies.[6]

| Cell | n kg U[a] | kg U[b] | Recovery kg U |
|---|---|---|---|
| A | 177 | 45 | 120 |
| B | 32 | 3 | 28 |
| C | 29 | 12 | 25 |
| isolated converter | 9 | 10 | 7 |

**Table 1:** Evaluation of PGDP Holdup Assay.

a. The neutron counters were not well collimated and measured an entire stage and double-counted the cooler.
b. Gamma-ray measurements covered only the converters.

Several bias estimates for gamma-ray holdup measurements have been reported**.** There is a stigma often associated with holdup uncertainty connected to the difficulty in obtaining results of quality similar to those obtained from NDA in well controlled situations. Many typical results from these difficult-to-measure situations are not publicly documented.[7] The following are summaries of gamma-ray-based nuclear material holdup measurements at multiple facilities. The percentages are the holdup results divided by reference values. The reference values are typically from measurements based on cleanout and recovery of the items measured for holdup.

- HEU processing 14% - 118%
- Pu processing 10% - 157%
- LEU processing 91% - 156%

The Rocky Flats Environmental Technology Site (RFETS) is located near Denver and contained 802 facilities. The Rocky Flats plant, which manufactured plutonium parts for nuclear weapons, was closed in 1989 and subjected to a 10-year cleanup campaign that ended in 2005 when RFETS was turned into a national wildlife refuge. During this period, $3.5 \times 10^5$ $m^2$ of buildings were dismantled and over 220 kg of plutonium holdup measured by a staff of 15. Holdup measurements included nearly 7 km of ductwork (~3 gPu/m), 1497 gloveboxes, and over 300 plutonium process tanks. Gamma-ray measurements were performed using HPGe and Bismuth Germanate detectors and the GGH procedures. All of the measured equipment was cleaned out and the recovery values can be compared with the pre- and post-cleanout NDA holdup measurements. The cleanout data were generally within 20% of the measured holdup. Some specific building values are listed in Table 2.[8]

| Building | Holdup/Recovered Pu |
|---|---|
| B-371 Gloveboxes | 1.09 |
| B-307 Ductwork | 1.06 |
| B-779 Total Holdup | 1.13 |
| B-A Total Holdup | 1.17 |
| B-B Total Holdup | 0.97 |
| B-C Total Holdup | 1.04 |
| B-D Total Holdup | 1.03 |

**Table 2:** RFETS Holdup Data.

A six-year study was conducted on the accuracy and precision of holdup measurements using the GGH (gamma ray assay) approach to measure simulated holdup situations with well known nuclear material standards. A series of simulations were fabricated for this study and a holdup training course; they included a pipe array, a steel pipe, an aluminum pipe, a rectangular ventilation duct, a V-Blender, and a contaminated spot on a floor. These were "salted" with U or Pu fuel rods, U metal foils, and small cans of $UO_2$ or $PuO_2$. Table 3 summarizes the results of this study which included measurements made by many people from students to holdup experts. The results shown here are "best case" vis-à-vis holdup assay accuracy.[9]

| | $^{235}U^{a}$ | $^{239}Pu^{a}$ |
|---|---|---|
| pipe array | 0.90 | 0.72 |
| V-blender | 1.22 | 1.02 |
| Al pipe | 1.03 | 0.97 |
| Steel pipe | 0.97 | 1.47 |
| floor spot | 0.96 | n/a |
| duct | 1.07 | 0.96 |

**Table 3:** GGH Holdup Assay Evaluation.

a. Number listed is the average ratio of measured U or Pu to the reference value.

A new holdup assay technique has been demonstrated in a uranium centrifuge enrichment plant and a MOX fuel fabrication facility. Distributed Source Term Analysis involves Monte Carlo modeling of the neutron field in a facility and sampling the actual neutron distribution with a portable neutron detector. While the data set is small, there are preliminary uncertainty estimates for DSTA:

> Plutonium – precision 20%, bias 150-400% measured high due to wrong source term
> Uranium – precision 4% (long counting times), bias 104% - source term was well known.
> For the DSTA approach to be successful, one must know well the chemical and isotopic composition of the deposits and it must be acceptable that this approach does not pin point deposit locations**.**[10]

## 7. Summary

- In general, holdup measurement uncertainties are larger than those for other NDA methods.
- Occasionally, with judicious use of cleanout and recovery, modeling, and data interpretation; uncertainties as good as 5% have been reported.
- The total amount of SNM measured is generally a monotonically increasing function of the number of holdup measurements. One should expect a decreasing rate of return after an initial modest effort when additional resources are applied to the measurements.
- The more time spent on a single measurement location, the better answer one expects for that measurement result, up to a point.
- The previous two truisms are not reliable:
  - What you don't know can hurt you.
  - Rely on cleanout or alternative measurements whenever possible.
  - Even experts get fooled.
- Additional funds might be best spent on cleaning out hot spots & comparing recovery to the holdup measurements to improve data collection procedure and analysis models.

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# Neptunium Flow-sheet Verification at Reprocessing Plants

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***Abstract:***

Due to their fissile nature, neptunium and americium have at least a theoretical potential application as nuclear explosives and their proliferation potential was considered by the IAEA in studies in the late 1990s. This work[1,2] was motivated by an increased awareness of the proliferation potential of americium and neptunium and a number of emerging projects in peaceful nuclear programmes which could result in an increase in the available quantities of these minor actinides. The studies culminated in proposals for various voluntary measures including the reporting of international transfers of separated americium and neptunium, declarations concerning the amount of separated neptunium and americium held by states and the application of flow-sheet verification to ensure that facilities capable of separating americium or neptunium are operated in a manner consistent with that declared.

This paper discusses the issue of neptunium flow-sheet verification in reprocessing plants. The proliferation potential of neptunium is first briefly discussed and then the chemistry of neptunium relevant to reprocessing plants described with a view to indicating a number of issues which relevant to the verification of neptunium flow-sheets. Finally, the scope of verification activities is discussed including analysis of process and engineering design information, plant monitoring and sampling and the potential application of containment and surveillance measures.

**Keywords:** neptunium, separation, partitioning, diversion, verification

## 1. Introduction

Neptunium-237 is the only significant isotope (in mass terms) of neptunium produced during irradiation of fuel in nuclear reactors. It has a critical mass of approximately 60 kg, not substantially greater than weapons grade uranium[3]. It is produced in nuclear reactors by three routes: i) from uranium-235 by successive neutron capture followed by beta decay; ii) by decay of Am-241 produced from uranium-238 by neutron capture and beta decay and iii) by (n,2n) reaction of uranium-238 and subsequent beta decay, i.e..

$$^{235}U \xrightarrow{n} {}^{236}U \xrightarrow{n} {}^{237}U \xrightarrow{\beta} {}^{237}Np$$

$$^{238}U \xrightarrow{n} {}^{239}U \xrightarrow{\beta} {}^{239}Np \xrightarrow{\beta} {}^{239}Pu \xrightarrow{n} {}^{240}Pu \xrightarrow{n} {}^{241}Pu \xrightarrow{\beta} {}^{241}Am \xrightarrow{\alpha} {}^{237}Np$$

$$^{238}U \xrightarrow{(n,2n)} {}^{237}U \xrightarrow{\beta} {}^{237}Np$$

Of these routes, the first is the most significant and thus it is seen that the amount of neptunium produced will be a function of the uranium enrichment of the fuel as well of course as the degree of irradiation. Typical irradiated light water reactor fuel (40 GWd/tU, 4% initial $^{235}$U) contains approximately 0.5 kg neptunium per tonne of fuel at discharge (approximately 5% of the amount of plutonium present) whereas irradiated natural uranium contains much smaller amounts, approximately

20 g $^{237}$Np per tonne U irradiated to 5 GWd/tU. Due to the long half-life of $^{237}$Np (2.1 x $10^6$ years) no significant reduction in mass occurs prior to reprocessing irrespective of the cooling period.

Certain properties of neptunium are compared to those of uranium and plutonium in Table 1. This shows that the properties of neptunium relevant to its potential use in nuclear explosives are not dissimilar to those of $^{235}$U, the principal difference lying in the significantly greater gamma dose arising from neptunium than from uranium. However, the dose from neptunium is not sufficiently large to pose a substantial difficulty in its handling and the lack of any appreciable spontaneous fission rate makes its use relatively attractive compared to plutonium, especially that containing an appreciable 240 isotopic fraction.

| Nuclide | Half –life (years) | Bare Critical Mass (kg) | Heat Emission Rate (Watts/kg) | Spontaneous Neutron Emission Rate (n/kg/s) | Gamma Ray Dose Rate (mSv/hr/kg at 1 cm) |
|---|---|---|---|---|---|
| $^{235}$U | 7.038 x $10^8$ | 50 | negligible | negligible | negligible |
| $^{237}$Np | 2.14 x $10^6$ | 60 | negligible | negligible | 1.04 |
| $^{239}$Pu | 24119 | 10 | 1.9 | 0.023 | 0.05 |
| $^{240}$Pu | 6564 | 40 | 7 | 1020 | 0.1 |
| $^{241}$Am | 433 | 70 | 114 | 1375 | 50 |

Table 1. Properties of $^{237}$Np compared to other fissile materials

Table 1 also includes information relating to americium 241. Although americium and neptunium are often considered together as alternative nuclear materials, it is clear from the properties presented here that the use of $^{241}$Am as a nuclear explosive faces much greater challenges than does that of $^{237}$Np due to its shorter half life which gives rise to significant radiation field and heat generation and also to the relatively high rate of spontaneous fission neutrons emitted. Whilst irradiated fuel contains comparable amounts of americium and neptunium (the americium content increasing with cooling time due to the decay of $^{241}$Pu) the greater difficulties associated with its handling result in it having a lower potential proliferation risk and thus it will not be considered further in this paper.

## 2. Behaviour of Neptunium in reprocessing plants

### 2.1 Neptunium chemistry

As with other elements, the routing of neptunium through a reprocessing plant depends upon its chemistry and in particular the relative affinity for the aqueous or organic phase of its different forms existing in the solvent extraction system. Neptunium lies between uranium and plutonium in the periodic table and in some respects its chemical behaviour is similar to these elements although there are significant differences. Like uranium and plutonium neptunium can exist in the +4 and +6 oxidation states as the $M^{4+}$ and $MO_2^{2+}$ cations (M=U, Np, Pu). These have a tendency to extract into the organic, TBP phase in the now universally adopted Purex process. However, unlike uranium and plutonium, the pentavalent form of neptunium is sufficiently stable towards disproportionation and oxidation by nitric acid to allow this species to exist for sufficiently long periods for it to influence the flow of neptunium through a reprocessing plant. For completeness, the trivalent form of neptunium, like that of uranium but unlike that of plutonium, is not stable in aqueous solutions and in any case would require a significantly stronger reducing agent to bring it about than is required for the corresponding reduction of plutonium. The chemistry of neptunium in the Purex process is therefore determined by processes controlling its interconversion between three chemical forms:

- Np(IV), $Np^{4+}$, extractable
- Np(V), $NpO_2^+$, inextractable
- Np(VI), $NpO_2^{2+}$, extractable.

The Purex process uses nitric acid as its aqueous base which as a result of thermally and radiolytically initiated reactions also contains nitrous acid ($HNO_2$). Nitric acid acts as an oxidising agent and thus has a tendency to oxidise neptunium towards Np(VI), however, nitrous acid acts as a reductant towards Np(VI) reducing it to Np(V).

Dissolution of irradiated fuel in hot, reasonably concentrated nitric acid converts most neptunium to the hexavalent form although subsequent conditioning of dissolver solution to ensure plutonium is in the optimum (tetravalent) form for the solvent extraction process also reduces Np(VI) to Np(V) along with the desired Pu(VI) to Pu(IV) conversion. The resulting Np(V) is however more easily reoxidised to Np(VI) than is Pu(IV) to Pu(VI) and thus the solution fed to the solvent extraction process of a reprocessing plant contains a mixture of extractable Np(V) and inextractable Np(VI) whereas plutonium is very largely in the tetravalent form with a very much smaller amount being present in the hexavalent form which, incidentally, is also extracted by the organic phase but to a lesser degree than the tetravalent form.

The behaviour of neptunium through the Purex process depends both upon the solvent extraction equipment used and the chemicals involved, particularly in the plutonium reduction step. In older reprocessing plants effectively all neptunium is extracted in the first solvent extraction cycle due to the use of mixer settlers having relatively long residence times. These allow the Np(V)/(VI) equilibrium to shift towards the right i.e. higher proportion of Np(VI) as a result of the extraction of nitrous acid into the organic phase followed by the oxidation of Np(V) in the aqueous phase. In more modern plants, using shorter residence time pulsed columns in place of mixer settlers, this oxidation step is incomplete with 66% of neptunium being reported to be extracted in the first step of the Thorp process[4] and 82% extracted in the UP3 process[5]. No published information is available concerning the Rokkasho Reprocessing Plant in Japan but its similarity with the French plant suggest that a similar routing to that observed in UP3 might be expected.

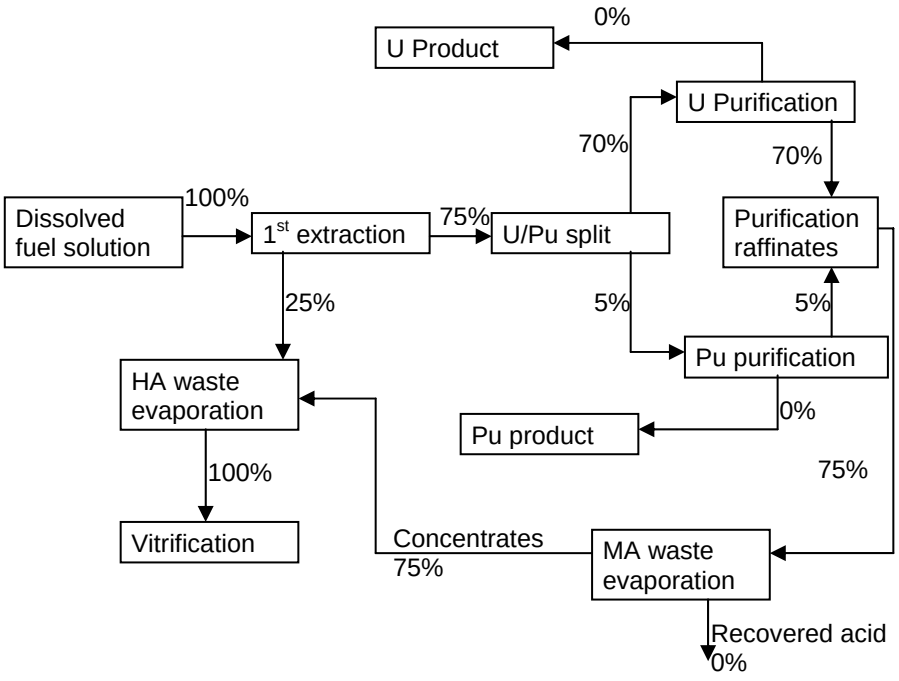


Figure 1. Nominal neptunium flows in a stylised modern flow-sheet

In these state of the art plants, neptunium very largely follows the uranium stream after the separation of uranium and plutonium as a result of the U(IV) reductant used being sufficiently powerful to reduce Np(V) and (VI) to the extractable tetravalent form which thus exits in the solvent phase and is backwashed along with the uranium and routed to the uranium purification cycle. Neptunium is separated very efficiently from the uranium product in the uranium purification cycle simply by a preconditioning step in which the acidity of the intermediate uranium product from the first solvent extraction cycle is increased and the liquor heated, oxidising neptunium present to Np(V). This inextractable neptunium then passes to the aqueous raffinate whilst the uranium extracts into the organic phase from which it is then backwashed using a low acid strip. Any small amounts of neptunium following the plutonium stream tend to be separated from the plutonium product during the plutonium purification cycle in which the conditioning step used to reoxidise plutonium to extractable Pu(IV) also conditions Np to the inextractable pentavalent state which then exists with the aqueous

raffinate leaving Pu in the solvent phase from which it is finally backwashed. A somewhat hypothetical and simplified flow-sheet of this modern type is shown in Figure 1. In this, the neptunium flows indicate 75% extraction in the initial cycle with all but 5% of this following the uranium path following the U/Pu split, for simplicity no Np is shown to reach either of the products and this is effectively true for the uranium product in reality whereas the plutonium product may contain up to a few percent of the total neptunium.

An alternative design of flow-sheet is shown in Figure 2, this is representative of certain older style flow-sheets[6,7]. The principal difference, in terms of neptunium routing, compared to Figure 1 is the relatively large proportion of neptunium which reaches the plutonium product. Such a flow-sheet is also relevant to future potential scenarios in which it is desired to minimise, or at least reduce, the amount of neptunium routed to vitrification in order to avoid issues associated with the final disposal of such wastes due to the very long-lived nature of the $^{237}$Np isotope and its mobility in the environment under oxidising conditions. Under such a scenario it may be preferential to route at least a proportion of the neptunium to the plutonium product for re-irradiation in a MOx fuel containing neptunium and americium, as this decays to give neptunium, in addition to uranium and plutonium.

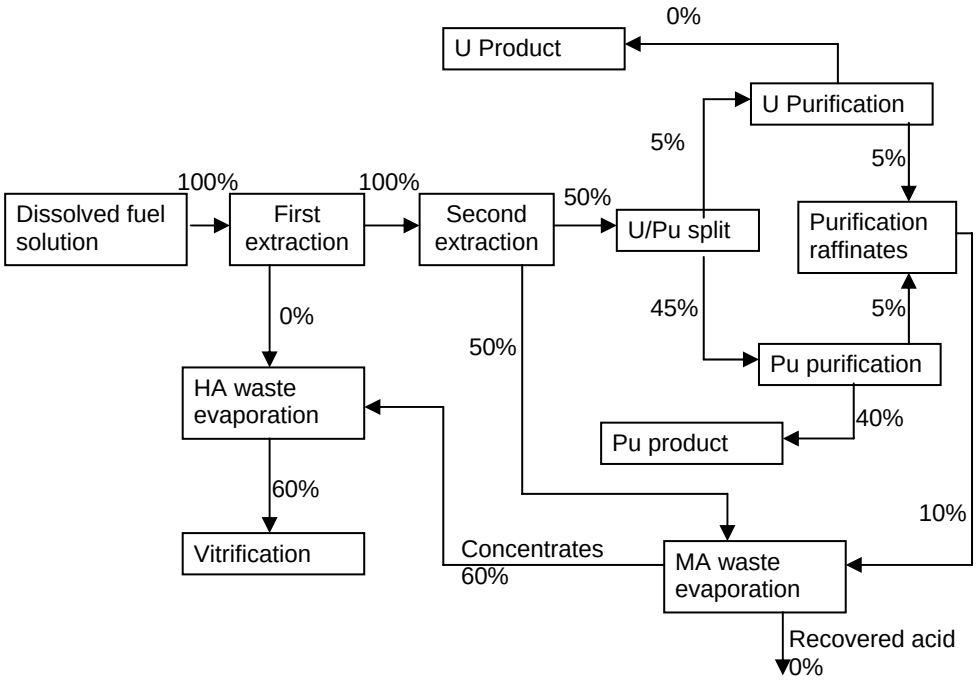


Figure 2. Nominal Np flows in an alternative flow-sheet

## 2.2. Diversion Analysis

In order to consider what activities are necessary to verify that a particular facility is being operated to its declared flow-sheet it is instructive to consider what potential re-routing options exist. Consider first the flow-sheet shown in Figure 1.

In this relatively simple flow-sheet the majority of neptunium present in the feed solution is extracted in the first solvent extraction cycle. As mentioned above, this proportion may be increased by increasing the residence time in these contactors which is not necessarily a trivial issue. Alternatively the oxidation of Np(V) to (VI) can be enhanced by a moderate increase in the acidity of the feed stream. Such studies have been undertaken in the framework of partitioning experiments designed to ensure recovery of neptunium for its re-irradiation in MOx[8]. Following this step the only significant flow of neptunium is with the uranium at the point of the U/Pu split and from there to the medium active aqueous raffinate. Whilst re-routing of, at least, a proportion of this neptunium to the plutonium stream is possible through use of either a less powerful reducing agent or controlled reoxidation, the simplest means of separating neptunium would be from the uranium purification process raffinate. Alternatively extraction could be from the combined purification cycle raffinates before or after evaporation. There would be some advantage in choosing to extract neptunium following the evaporation step as the volume of liquor to be treated would be significantly less than before the evaporation step. Extraction

of the relatively small amount of neptunium routed to the first cycle raffinate is unattractive due to the presence of the majority of the fission products in this stream which is therefore intensely radioactive.

The alternative flow-sheet shown in Figure 2 is more complicated and offers more opportunities for neptunium to be re-routed, these can be summarised as follows:

1) Divert a proportion of neptunium with the aqueous raffinate from the first solvent extraction cycle
2) Extract less Np in second extraction cycle
3) Extract more Np in the second solvent extraction cycle
4) Route more neptunium with the uranium stream following the U/Pu split stage and separate the neptunium in the uranium purification cycle in a similar fashion to that shown in Figure 1.
5) Increase the separation of neptunium from plutonium in the plutonium purification process

Consider then what would be required to bring about these changes:

1) To route more Np with the aqueous raffinate from the first cycle 1 requires either a reduction in the residence time, a reduction in solvent to aqueous flow ratio or acidity or addition of a relatively weak reducing agent to the aqueous scrub. Of these, changes in acidity, solvent:aqueous flow rate or residence time are likely to result in unacceptable changes (losses) in plutonium behaviour. Use of a mild reducing agent in the aqueous strip could potentially be used but would be difficult to implement in such a complicated system. Recovery of Np from the raffinate would require its oxidation followed by extraction to separate it from the fission products followed by backwashing. As all the fission products would also be present, at least during the extraction stage, full shielding would be required. This is a relatively unattractive option.

2) Decreasing the extraction of neptunium in the second cycle would require the addition of a mild reducing agent in order to reduce Np(VI) to Np(V), a source of nitrous acid is one possibility whilst reducing the acidity of theses stages might conceivably be used to extract less neptunium but it is likely that this would be accompanied by significant losses of plutonium to the aqueous raffinate. Recovery of neptunium diverted to the aqueous stream at this point would be simpler than in the case above due to the reduction in fission product activity. This is a moderately unattractive option.

3) The first and second solvent extraction cycles of this flow-sheet are reasonably similar and there is therefore some potential to increase the proportion of neptunium extracted at this stage however this may require changes to the flow-sheet which have a deleterious effect on plutonium extraction and so would be difficult to introduce in practice.

4) Extraction of additional neptunium with the uranium in the U/Pu split stage is probably the most simple way in which the flow of neptunium could be re-routed in the whole flow-sheet as this is the route taken by the majority of the neptunium in the flow-sheet shown in Figure 1. Additional neptunium routed this way would be backwashed with the uranium and enter with the aqueous feed to the uranium purification process during which it could be separated from the uranium by controlled oxidation e.g. by use of moderate acidity and temperature in a pre-conditioning step to oxidise Np(IV) to (V). Thus the neptunium would be directed to the aqueous raffinate and, if not diverted, combined with other raffinates in the MA evaporator.

5) To prevent neptunium extracting in the plutonium purification process would probably require its conversion or stabilisation in the pentavalent form. As has been suggested above, treatment with nitrous acid may potentially be used to accomplish this.

The ease with which neptunium could be recovered from different streams depends upon a number of factors including.

- throughput and concentration of neptunium;
- fission product concentration;
- difficulty of separation chemistry; and
- engineering simplicity of diversion or concealment.

These factors suggest that, the most attractive streams from a neptunium diversion perspective are, in the case of the flow-sheet shown in Figure 1,

- the aqueous raffinate from the uranium purification process or any latter path taken by this stream;

and in the case of the flow-sheet shown in Figure 2:

- either the aqueous raffinate from the second solvent extraction cycle, concentrates from the medium active evaporator or potentially the aqueous raffinate from the plutonium purification cycle.

## 3. Flow-sheet Verification

The approach to neptunium flow-sheet verification at reprocessing plants adopted by the Agency involves a set of measures to provide assurance that undeclared recovery or separation of neptunium is not performed. Implicit in this approach is the requirement that safeguards measures are applied to plutonium and uranium which the plant is designed to recover. The starting point for verification is then based upon a declaration by the state concerned as to the distribution of neptunium through the process.

This declaration and other information such as facility layout and process description provided in connection with the application of safeguards to uranium and plutonium at a reprocessing plant are then used to determine what additional activities are required in connection with neptunium flow-sheet verification. In general terms the following activities may be considered:

- examination and verification of facility design information, in particular as it relates to the processing of neptunium;
- examination of relevant process records, supplemented by monitoring key process parameters to check consistency with declared operation;
- measurements to confirm that the actual flows and distributions of neptunium are broadly consistent with the declared flow-sheet;
- application of containment and surveillance measures; and
- use of environmental sampling and analysis.

These are considered in more detail below.

### 3.1. Design Information Examination and Verification

Design information examination and verification is a key component of the flow-sheet verification approach and is an extension of the activities undertaken in reviewing facilities in the course of safeguarding uranium and plutonium. However, in this application greater attention is focussed on areas such as raffinate streams containing neptunium which may contain trivial amounts of uranium or plutonium.

The activity essentially involves examination of the information provided by a state concerning a facility's design, purpose, throughput and flow-sheet to identify expected routings of neptunium and to confirm that processes to recover neptunium are not included as part of the design. The behaviour of neptunium in two conceptual flow-sheets and its potential re-routing have been discussed above in section 2, such analysis may be refined for specific plant conditions or when more details of chemical flow-sheet or process equipment are known.

The physical status of the plant is also considered against the declared design to ensure for example that empty cells, shielded spaces and duplicate or redundant equipment are not used for recovery of neptunium. Similarly, where possible, pipe-work arrangements may be analysed to allow potential by-pass routes or take-off points to be identified and in addition the plant is examined to confirm that the as-built condition accords with the design and that additional features relevant to neptunium separation have not been incorporated.

### 3.2. Examination of Process Records and Monitoring of Key Process Parameters

Flow-sheet verification relating to neptunium is based on the premise that the reprocessing facility concerned is designed and operated to recover only uranium and plutonium. As part of flow-sheet verification, examination of process records relating to plant operation may be extended to look for evidence such as abnormal flows or treatments, such as excessive re-work, which may be indicators of the possible separation of neptunium. Additional monitoring of key process parameters relevant to

neptunium flows may also be undertaken, for example those relevant to medium active raffinates which generally contain the highest concentrations of neptunium (see above in Section 2)

### 3.3 Confirmatory measures

Significant plant sampling is generally undertaken at reprocessing plants in respect of safeguards activities associated with uranium and plutonium, such samples may also be analysed as part of flow-sheet verification to ascertain that the distribution of neptunium throughout the plant is consistent with that declared. The objective here is not to build a fully quantitative description of neptunium routing but merely to confirm that the major flows are broadly consistent with those expected from the information provided to the Agency. However, whilst the use of samples taken for uranium and/or plutonium accountancy purposes may give a reasonable indication of neptunium flows under certain circumstances there are instances such as in the case of the flow-sheet depicted in Figure 1 in which the amount of neptunium reaching the purified products is very small indeed. In such cases it may be desirable to take samples of intermediate streams, or of raffinate flows containing relatively high neptunium concentrations but with only small amounts of uranium or plutonium. By inclusion of highly active waste in storage and the feed of this to a vitrification process in the list of sampling streams it is possible to gain assurance that neptunium is not being separated following the reprocessing process and is being converted to a waste form from which its recovery is essentially impractical,

These samples may be analysed for the absolute concentration of neptunium, or for the ratio of neptunium present compared to a marker with known distribution through the plant. One example would be to compare the amount of neptunium present in the dissolver solution to that of uranium, plutonium and a fission product such as neodymium. Ratios of these elements can then be measured at later stages in the process e.g. highly active raffinate and uranium and plutonium product solutions in order to determine the proportional distribution of neptunium throughout a facility.

### 3.3 Containment and Surveillance

Design information examination and verification should identify potential removal routes and take-off points for neptunium. Some of these may already be covered by containment or surveillance measures put in place with regard to safeguards relating to uranium or plutonium. Additional measures specifically relating to neptunium may, however, be desirable and could be introduced with the agreement of the state, noting the voluntary nature of neptunium flow-sheet verification activities.

### 3.4 Environment Sampling and Analysis

In principle, both bulk and particle analysis of environmental samples may be used for the detection of neptunium although with all such sensitive analysis results need to be interpreted with care. Possible indicators of undeclared neptunium separation may include:

- neptunium to plutonium ratios in product material or intermediate streams inconsistent with declared flow-sheets; or
- neptunium in a chemical form or degree of purity inconsistent with declared activities.

## 4. Conclusions

Commercial scale reprocessing operations involve the processing of substantial amounts of neptunium (of the order of 500 kg per year for a modern plant). Such plants are designed and operated such that the majority of the neptunium is routed eventually to vitrification along with the fission products and other minor actinides. However because of its chemical nature the routing of neptunium is susceptible to alteration through relatively small process or equipment changes. Alternatively, as the majority of the neptunium present is found in the feed to and product from the medium active, or acid recovery, evaporator such a stream may potentially be diverted and the neptunium content extracted elsewhere.

To combat the potential diversion of neptunium, the concept of flow-sheet verification has been developed. As has been described above, whilst this does not apply the same quantitative level of

measures as are applied to uranium and plutonium accountancy, it nevertheless offers a means for ensuring that plant operation with respect to neptunium accords with that declared.

Measures such as flow-sheet analysis and physical design information verification coupled with plant sampling and other activities are used to ensure that neptunium flows are consistent with those expected and to give a high degree of confidence that undeclared neptunium separation or diversion is not occurring. Ultimate reassurance is obtained once it can be shown that the expected quantity of neptunium is combined with fission products in the highly active waste converted to a vitrified waste product.

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# "InSpect" – a measuring technique and a software complex for gamma-activity analysis of radioactive wastes and contaminations

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***Abstract:***

*Software complex "InSpect" was developed for measuring of radionuclide concentrations in the nuclear-power industry wastes, such as metal drums of about 200-litres in volume, reinforced concrete cylinder and rectangular containers of up to several cubic meters in volume, etc.*

*Technique advantages:*

- *It is easy adaptable to various up-to-date spectrometric equipment with a high-resolution Ge-detector;*
- *It allows to perform measurement of various shapes and configuration containers, and it is easy adaptable to new geometries;*
- *It allows to measure containers' activity on the assumption of non-uniform activity volumetric distribution which can vary from $10^2$ to $10^{10}$ Bq /litre*

*Technique features:*

*The flux density of the volumetric source in the specific measurement geometry is calculated by double integration. For large volumetric sources with strong absorption the integration is performed not for the whole volume, but only for the part whose radiation is registered by the detector. "InSpect" displays "radiating" volume of the space source to understand what portion of the information about a radionuclide concentration in the container is possible to obtain and what measurement strategy to choose. If the waste in the container has a heterogeneous structure or the container has a complex geometry, it is necessary to perform several measurements at different viewing angles and heights.*

*Our measuring technique was certificated in 2005 at the All-Russian Institute of Physical-Technical and Radiotechnical Measurements of the Russian Agency for Standards. Since the December of 2005 as a part of spectrometric equipment "Sadovnik" produced by RPE "Doza", software complex "InSpect" has been used at the Moscow Scientific and Industrial Association "Radon" for controlling of municipal radioactive waste. The English version of "InSpect" as a part of "Sadovnik" was delivered to the VINCA Institute of Nuclear Science of Serbia and Montenegro in September 2006.*

*The results of the "InSpect" tests and verification will be given at the presentation.*

**Keywords:** gamma-activity; radioactive; RadWastes; software; spectrometer

## 1. Introduction

The Russian Federation sanitary regulations [1] include the control of specific activity and radionuclide composition of radioactive waste (RadWastes) to guarantee safe storage and management of RadWastes. The object of such control at nuclear-power enterprises are usually RadWastes containers of various shapes and configurations, especially jerricans, drums, nonrecoverable shielding containers, tanks with liquid or solid RadWastes, scrap-metal or demolition wastes in boxes, tatters packing, etc. Specific activity varies within a wide range from hundreds of Bq/litre to $10^{10}$ Bq/litre and even more.

Most man-caused and natural radioactive nuclides in at RadWastes, which are to be controlled, can be characterized by the gamma-radiation with energy from several tens of KeV to several MeV. This fact predetermines the selection of the high-resolution gamma- spectrometry with Ge-detectors as the main instrument of activity control and radionuclide composition control of RadWastes, though in some cases it is possible to use scintillation detectors

The RPE "Radiation control. Methods and equipment" developed a software complex "InSpect" for controlling the RadWastes' activity with the help of spectrometric measuring systems. This complex measures initial and conditioned RadWastes with unknown radionuclide composition directly at containers without their opening; it also sorts RadWastes according to specific activity's categories. Software complex "InSpect" makes it possible to:

- determine specific activity and full activity of RadWastes containers;
- determine radionuclide composition of container's contents, including specific activity of each nuclide identified;
- RadWastes classify according to specific activity categories;
- perform container's certification, i.e. to organize a printing report of processing results according to operating enterprise's requisitions;
- save the measuring results to the data base (including intermediate processing results and the spectra themselves) with the ability to sanctioned access through local network.

## 2. Physical principles

The following physical principles formed the basis of software complex "InSpect". Unlike traditional methods of radiation transport calculation, when co-ordinates are connected to the source, our method uses spherical co-ordinates connected to the detector. This gives us a possibility to speed up a calculation process, by restricting the source to a "visible" (radiating) volume, and to simplify the analysis of radiation transport for the sources of complex shape geometry.

Counting rate $N$ at the full absorption peak can be expressed as an integral of intensities, which are spaced through the source volume with the volume density of emitters $q$, and which are normalized by the flux density:

$$N = \int_v \frac{q}{4\pi r^2} P(\vec{r}) dV , \qquad (1)$$

where $P(\vec{r})$ is the probability of radiation, which developed appeared in the volume element $dV$, to be registered at the full absorption peak.

After turning to the spherical coordinate system connected with cylindrical detector (see Fig. 1), the equation is transformed to the following form:

$$N = \frac{q}{4\pi} \int_\varphi d\varphi \int_\vartheta \sin\vartheta d\vartheta \int_r P(\varphi, r, \vartheta) dr , \qquad (2)$$

where $N$ is the counting rate at the full absorption peak, $q$ is the volume density of emitters in the source, $\varphi$ is the azimuth angle, $\vartheta$ is the polar angle, $r$ is the radius-vector from the detection point to

the element dV , $P(\varphi,r,\vartheta)$ is the probability of radiation, which developed in the volume element dV , to be registered at the full absorption peak.

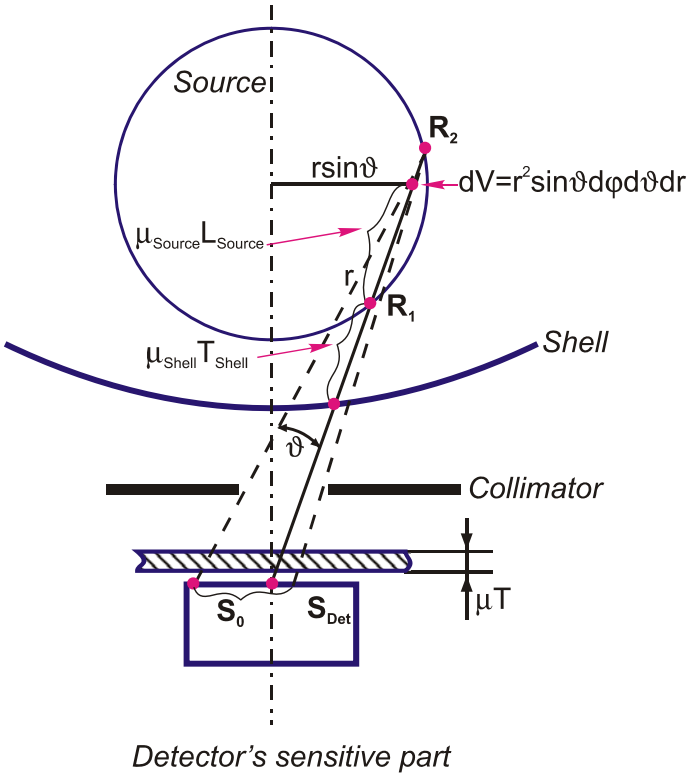


**Figure 1:** Radiation transport from the source to the collimated detector.

By assuming that the source (container's matrix with spaced radionuclides) is homogeneous and singly connected (a ray from the detection point crosses the source at the two points: $R_1$ and $R_2$ (see Fig. 1)), one can integrate (1) over r and separate the probabilities with the absorption in the container and detector:

$$N = \frac{q}{4\pi\,\mu_{Source}} \int_{\varphi} d\varphi \int_{\vartheta} (P_0(\vartheta) \cdot \sin\vartheta \cdot \Psi(\varphi,\vartheta,\mu))\,d\vartheta\,, \qquad (3)$$

$\Psi(\varphi,\vartheta,\mu)$ is the probability of radiation to pass through the source and its container's shells:

$$\Psi(\varphi,\vartheta,\mu) = \exp\left(-\mu_{Shell} L_{Shell}(\varphi,\vartheta)\right) \cdot \left\{1 - \exp\left[-\mu_{Source}\left(R_2(\varphi,\vartheta) - R_1(\varphi,\vartheta)\right)\right]\right\} \qquad (4)$$

where $R_2$ , $R_1$ are the points, where a ray from the detection point crosses the emitting volume of the source, $L_{Shell}$ is the route length in the container's shell; $\mu_{Shell}, \mu_{Source}$ are the linear coefficients of radiation reduction in the shell and in the source, respectively;

$P_0(\vartheta)$ is the probability of registration at the full absorption peak for collimated detector, when radiation falls on the detector at the $\vartheta$ angle.

$$P_0(\vartheta) = \left[\left(\varepsilon \cdot S_{Det} \cdot P_{Ph}\right)\right] \cdot \left[f(\vartheta) \cdot \cos\vartheta \cdot \exp\left(-\mu T/\cos\vartheta\right)\right] \qquad (5)$$

where ε is the registration probability of the given energy radiation at $\vartheta = 0$; $S_{Det}$ is the open detector area; $P_{Ph}$ is the photographic contribution for the given energy and given detector at $\vartheta = 0$; $f(\vartheta)$ is the share of the "illuminated" detector's surface, not shaded with collimator, $f(\vartheta) = S_0 / S_{Det}$; μT is the effective thickness of the detector's case and its dead layer.

In is clear from (5), that $P_0(\vartheta)$ consists of two functionals, one of which is an energy dependence of radiation registration effectiveness of the detector at $\vartheta = 0$, and the other is an angular detector's sensitivity, which does not depend on energy, the both functionals being experimentally determined.

Figure 2 shows a graph of the product approximation $(\varepsilon \cdot S_{Det} \cdot P_{Ph})$. The coefficients of the energy dependence approximation of radiation registration probability at the full absorption peak are determined experimentally from a series of measurements of standard sample point sources with a representative number of gamma-lines; these sources are placed at a fixed distance along the detector's axis.

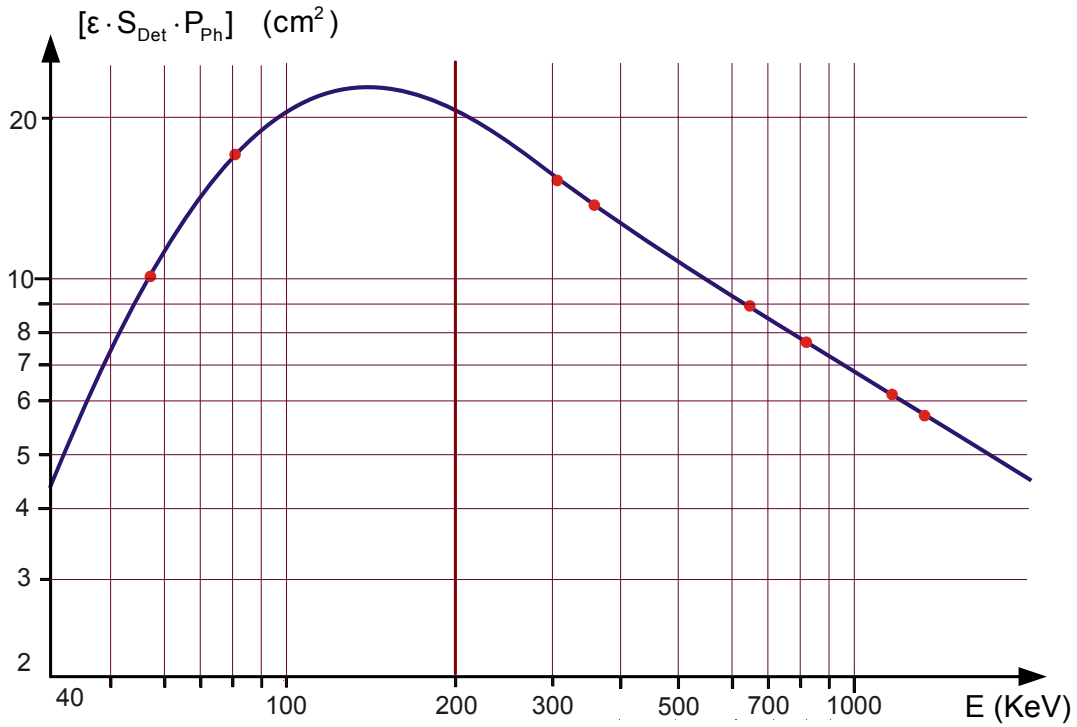


**Figure 2:** Sample of energy dependence approximation of radiation registration probability at the full absorption peak (logarithmic scale)

The angular dependence $f(\vartheta)$ can also be determined experimentally from a series of measurements of standard sample point sources placed at a fixed distance at various angles to the detector's axis.

Having performed a series of measurements with step-by-step angle $\vartheta$ changing for specific system "detector + collimator", one can receive a set of values located on a surface in the "probability – angle - energy" space. Then one can determine an angular dependence approximation of registration probability at the full absorption peak at fixed energies. Solid lines in Fig. 3 show calculation results of $f(\vartheta)$ for energies 100 KeV and 1 MeV. Different points in Fig. 3 show the results of measurements with $^{133}$Ba, $^{137}$Cs, $^{60}$Co, $^{152}$Eu sources. Also a fluctuation range is shown for $P = 0.95$ of measurement results for initial and final angles. From the figure one can see that the shape of the angle function approximation is close (similar) for substantially differentiating gamma-quanta energies. So, within the

limits of experiment's inaccuracy, the angle function approximation can be determined by a unified approximation line within the entire energy range.

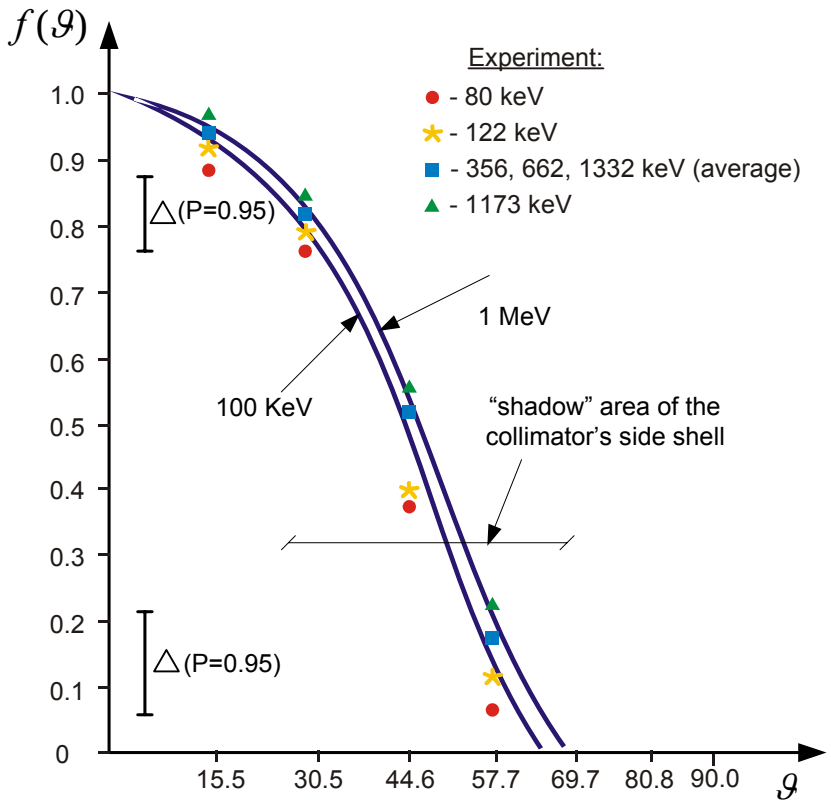


**Figure 3:** Angle function $f(\vartheta)$ for collimator with 30° span.

One more useful feature of our technique is a calculation of the so-called "visible" (radiating) volume, which allows changing integration over the whole source volume for integration over a specific surface while performing calculation of radiation passing through the source. This surface is determined from the following.

Taking into account the analytical integration on r coordinate in formula (3) for counting rate at the full absorption peak, it is possible to find the $R_2^{\beta}(\vartheta,\varphi)$ point in any ray inside the source, so that the integral over r from $R_1(\vartheta,\varphi)$ to $R_2^{\beta}(\vartheta,\varphi)$ becomes part of β in the integral over r from $R_1(\vartheta,\varphi)$ to $R_2(\vartheta,\varphi)$ :

$$R_2^{\beta}(\vartheta,\varphi) = R_1(\vartheta,\varphi) + \frac{1}{\mu_{Source}} \ln\left[\frac{1}{1-\beta\left\{1-\exp\left[-\mu_{Source}\cdot\left(R_2(\vartheta,\varphi)-R_1(\vartheta,\varphi)\right)\right]\right\}}\right] \qquad (6)$$

In case of the source section at the given angle value $\varphi$, for example, $\varphi = 0$ or $\varphi = \frac{\pi}{2}$, the dependence line $R_2^{\beta}(\vartheta,\varphi)$ for the given β value, for example, $\beta = 0.95$, can be drawn on this section. The similar dependences are given in Fig. 4 for cylindrical and rectangular sources.

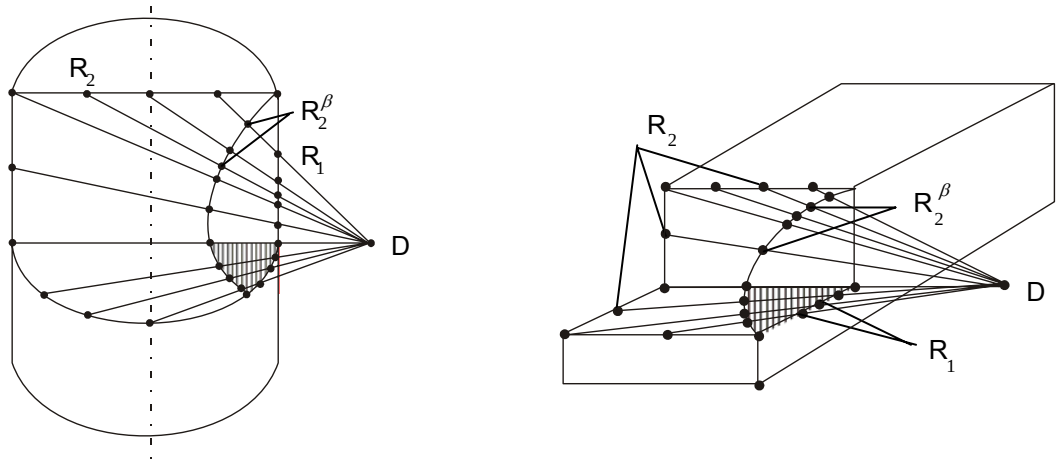


**Figure 4:** $R_2^{\beta}(\vartheta,\varphi)$ dependence at the given $\varphi$ value for cylindrical and rectangular sources

Hence, having expanded equation (3) in independent components, and having restricted the integration space to the "visible" volume, we performed analytical integration and received the mathematical tool, which can quickly and with adequate accuracy calculate gamma-radiation efficiency in any system like "measuring object + collimator + detector" with the help of simple analytical energy and angle approximations of detector's efficiency, which were obtained on the basis of experimental point-source measurement data.

## 3. Test and validation results

Software complex "InSpect" was tested at several stages

At first measurements of the RadWastes container were made at Smolensk Nuclear Power Plant.
A standard 200-litre steel drum filled with middle-active liquid RadWastes was used as a measurement object. Specific activity of RadWastes and radionuclide contents in the drum were determined by sample cutting analysis method. The results of laboratory analysis of sample cutting are shown in Table 1.

| | $^{137}$Cs | $^{134}$Cs | $^{54}$Mn | $^{60}$Co | $^{58}$Co | $^{95}$Nb | **Full activity** |
|---|---|---|---|---|---|---|---|
| **Sample activity (200 ml), MBq/litre** | 70.0 | 34.0 | 3.55 | 1.65 | 0.500 | 0.185 | 2197 **MBq** |
| **Ratio uncertainty, %** | 11 | 11 | 12 | 15 | 17 | 20 | 12 |
| **Contribution portion to activity, %** | 63.0 | 30.8 | 3.2 | 1.5 | 0.5 | 0.2 | 100 |

**Table 1:** Results of sample cutting analysis

Gamma-spectra collection from the drum filled with liquid RadWastes was performed with the help of spectrometry plant "ISOCS" [2] of "Canberra Industries", which is designed for activity measurements and analysis of RadWastes' radionuclide contents in containers of various shapes and geometry. Gamma-spectra measurements of the drum were performed at several distances between the drum and the collimator's cut: 520 mm, 750 mm, and 1140 mm. Then the spectra were analyzed by

software complex “ISOCS” of “Canberra Industries” and our software complex “InSpect”. The processing results obtained were compared with the sample cutting analysis; these are shown in tables 2 and 3 as a ratio of measured specific activity to sample specific activity.

| Ratio of measured activity to sample activity | $^{137}$Cs | $^{134}$Cs | $^{54}$Mn | $^{60}$Co | $^{58}$Co | $^{95}$Nb | Ratio of full activity |
|---|---|---|---|---|---|---|---|
| at 520-mm distance from drum | 1.20 | 1.29 | 1.09 | 0.95 | 1.27 | 0.67 | 1.210 |
| at 750-mm distance from drum | 1.19 | 1.27 | 1.07 | 1.00 | 1.16 | 0.54 | 1.196 |
| at 1140-mm distance from drum | 1.21 | 1.25 | 1.07 | 1.09 | 1.23 | 0.68 | 1.205 |

**Table 2:** Results of “ISOCS” analysis

| Ratio of measured activity to sample activity | $^{137}$Cs | $^{134}$Cs | $^{54}$Mn | $^{60}$Co | $^{58}$Co | $^{95}$Nb | Ratio of full activity |
|---|---|---|---|---|---|---|---|
| at 520-mm distance from drum | 0.80 | 0.87 | 0.77 | 0.70 | 0.79 | 1.31 | 0.813 |
| at 750-mm distance from drum | 0.81 | 0.87 | 0.77 | 0.75 | 0.84 | 1.44 | 0.821 |
| at 1140-mm distance from drum | 0.85 | 0.90 | 0.80 | 0.84 | 0.88 | 1.36 | 0.858 |

**Table 3:** Results of “InSpect” analysis

It is clear that basically, the “ISOCS” analysis gave overestimated results, and “InSpect”, underestimated results. Since 95% of contribution was provided by $Cs^{137}$ and $Cs^{134}$ and taking into consideration uncertainty of sample cutting measurements and uncertainties of drum measurements, which constitute about the same values, the testing results should be considered satisfactory.

At the nest stage the software complex “InSpect” was tested at the Moscow Scientific and Industrial Association “Radon” as a part of spectrometric equipment “Sadovnik” [3] produced by RPE “Doza” (with “Ortec’s” [4] detector). Five-litre rectangular jerricans, 200-litre drums and nonrecoverable shielding concrete containers comprising four drums were used in measurements. Liquid RadWastes with predetermined chemical composition, density and specific activity were used to prepare measurement objects. Also the measurements of real containers with known contents (e.g., scrap-metal, demolition wastes or cables) were made.

As a case in point here are the results of the standard 200-litre drum measurements. That drum had several layers with different specific activity of RadWastes, as it was filled up with 70-mm of pure cement. After induration of the concrete, the drum was filled up with liquid RadWastes mixed with cement and claydite to 1/3 of the drum’s volume. After induration of the second layer the third layer was added. So, we prepared the drum with irregularly dense and active layers. Specific activity of RadWastes and radionuclide drum’s contents were determined with the help of sample cutting analysis method before the drum matrix was formed. The results of laboratory analysis of sample cutting are shown in Table 4.

| | $^{137}$Cs | $^{134}$Cs | $^{60}$Co | Full activity |
|---|---|---|---|---|
| **Sample activity (200 ml), Bq/litre** | $1.95 \cdot 10^5$ | $2.96 \cdot 10^3$ | $8.25 \cdot 10^2$ | $3.98 \cdot 10^7$ **Bq** |
| **Ratio uncertainty, %** | 15 | 9 | 13 | 15 |
| **Contribution portion to activity, %** | 98.1 | 1.5 | 0.4 | 100 |

**Table 4:** Results of sample cutting analysis

The rotating drum was measured at the heights of 140 mm, 420 mm and 700 mm from drum's bottom with the help of software complex "InSpect". The results are shown in Table 5 as a ratio of the measured specific activity to the sample specific activity.

| Ratio of measured activity to sample activity | $^{137}$Cs | $^{134}$Cs | $^{60}$Co | Ratio of full activity |
|---|---|---|---|---|
| **at 140-mm height of drum's bottom** | 0.86 | 0.78 | 0.88 | 0.86 |
| **at 420-mm height of drum's bottom** | 0.90 | 0.93 | 0.91 | 0.90 |
| **at 700-mm height of drum's bottom** | 1.24 | 1.18 | 1.22 | 1.23 |
| **Average by height** | 1.05 | 1.00 | 1.07 | 1.05 |

**Table 5:** Results of "InSpect" analysis of heterogeneous RadWastes

Thus, the tests carried out at Smolensk Nuclear Power Plant and at the Moscow Scientific and Industrial Association "Radon" showed that software complex "InSpect" gives satisfactory results and can be used in routine measurements of radioactive wastes.

## 4. Conclusions

As a conclusion we would like to note that:

1. Software complex "InSpect" was developed by the RPE "Radiation Control. Methods and Equipment" to perform routine measurements of containers with RadWastes.

2. "InSpect" has a friendly user interface.

3. "InSpect" is easily adaptable to various up-to-date spectrometric equipment. It can be used as a part of measurement plant, such as "ISO-CART" of "Ortec", "ISOCS" of "Canberra Industries", "Sadovnik" of RPE "Doza" and any other plants with a high-resolution Ge-detector and modern spectrometry route.

4. "InSpect" has a series of predefined templates for measuring containers of various forms.

5. It is enough to perform one calibration of energy/angle efficiency of system “detector + collimator” to perform serial measurements of containers of various forms.

6. Experimental tests and verification of “InSpect” showed satisfactory results, which made it quite applicable to routine measurements of radioactive wastes.

7. “InSpect” is used at the Moscow Scientific and Industrial Association “Radon” (the Russian Federation) and at the VINCA Institute of Nuclear Science (Serbia and Montenegro).

8. At the moment we are delivering “InSpect” to Volgodonsk Nuclear Power Plant and Novovoronezh Nuclear Power Plant developing our own multi-functional plant for RadWastes certification.

## 4. Acknowledgements

We would like to express our thanks to Smolensk Nuclear Power Plant for their help in organizing tests of our software complex “InSpect”.

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# A NEW EVALUATION OF THE HALF-LIFE OF $^{241}$PU

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***Abstract:***

Some of the longest and most accurate measurements on the half-life of plutonium-241 have been carried out at IRMM. The measurements started in 1976 and were all made on a single sample of highly enriched $^{241}$Pu.
A double isotopic ratio methods is applied in which $n(^{241}Pu)/n(^{240}Pu)$ is compared with $n(^{240}Pu)/n(^{239}Pu)$ to yield an effective ratio $n(^{241}Pu)/n(^{239}Pu)$. This method is internally highly consistent and repeated measurements on the same solution have yielded the value of 14.29 years [1].
The last measurements were carried out in 1996 and therefore it was considered that this was an appropriate time to carry out another repeat on this solution.
Previous measurements were done on a standard thermal ionisation mass-spectrometer (TIMS) using a single Faraday collector in the peak jumping mode. For the present set of measurements a different TIMS was also used in which a completely different analysis type was applied: total evaporation with multi-Faraday collectors.
The accumulated set of measurements includes work done using several mass-spectrometers and operators. Great care was taken throughout for each measurement to eliminate $^{241}$Am from the plutonium solution before preparing the filaments for mass-spectrometry.
An overview of the past and present measurements will be given and the final value from the measurements will be discussed.

***Keywords:*** Half-life; plutonium-241;mass-spectrometry

## Introduction

Because $^{241}$Pu has the shortest half-life of the plutonium isotopes commonly encountered in material recovered from irradiated nuclear fuel, it affects the measurement of the amount of plutonium for accountancy purposes to a much greater extent than other isotopes. Corrections for decay of this isotope are often the largest contributor to overall uncertainties in the total amount of plutonium, especially after storage for several years.
Many laboratories have measured the half-life of this isotope and an overview was given in an earlier paper [1]. Values lay in the region between 13.0 to 13.5 years. This spread was considered unacceptable and in 1977, in an attempt to measure the half-life using the best technique possible a series of isotope ratio measurements on an plutonium sample enriched in $^{241}$Pu were started at the reference materials institute of the European Commission, then called 'CBNM' and now the 'Institute for Reference Materials and Measurements, IRMM'.
The strength of this measurement lay not only in the being able to apply the power of thermal ionisation mass-spectrometry, but further in the design of the measurements for which a ratio-of ratio technique was used to eliminate, at least to the first order, mass bias effects.
Results for measurements carried out over the 20 years to 1997 were published previously [1]. The measurements gave an excellent fit to a decay with a resulting half-life of 14.290(6) years, a value lower than previously measured but also with an uncertainty lower than previously achieved.
The original material is still carefully kept at IRMM and was available for further measurements. The

first measurements were made (in 1977 and later) using a single-collector TIMS and the measurements made in 1997 were also made using this technique, although on a different instrument. Nowadays, isotope ratios of Pu are preferentially carried out on a multi-collector Faraday detector array applying the technique of total evaporation in which a smaller sized sample (typically between 50 and 500 ng Pu) is evaporated to completion. This method essentially removes mass fraction ('mass bias') effects from the isotope ratio measurements.
Over the last 10 years the impact of the Guide to the Expression of Uncertainty in Measurements (GUM) published by BIPM/ISO [2] has had an increasing impact on the calculation of uncertainties in the nuclear area. It appeared to us that the time was ripe to remeasure the isotope ratios of the material originally used in 1977 applying also the alternative measurement technique of total evaporation combined with a multi-collector Faraday array and to use the opportunity to apply as rigorously as possible the concepts of the GUM by means of a dedicated software [3] to arrive at a value of the half-life and associated uncertainty that would be regarded as definitive by the world nuclear measurement community.
Moreover, during the course of the present measurements and calculations, the calculations from 1997 were found to inadvertently have included a double correction for the decay of the isotopes, $^{240}$Pu and $^{239}$Pu, which lowered the value of the half-life of $^{241}$Pu. The opportunity was taken to therefore calculate a new value of the half-life and at the same time to critically look at its uncertainty estimation and re-evaluate this as well.

## New Measurements

The original material was supplied from Oak Ridge. The original and present isotopic composition of the sample is given in Table1. The remainder, about 7.4 mg of plutonium has been carefully stored in a glass bottle behind lead shielding because of the heavy in-growth of $^{241}$Am.

| Isotope | 100 x isotope amount fraction | |
|---|---|---|
| | 1976 Jan 15 | 2006 Dec 1 |
| $^{238}$Pu | 0.01 | 0.03 |
| $^{239}$Pu | 1.15 | 4.10 |
| $^{240}$Pu | 4.05 | 14.42 |
| $^{241}$Pu | 92.75 | 74.09 |
| $^{242}$Pu | 2.06 | 7.36 |

Table 1: Isotopic composition of $^{241}$Pu base material expressed as $n(^{x}\mathrm{Pu})/n(\mathrm{Pu})$

### Chemical preparation of sample

It is critical in these measurements to remove $^{241}$Am completely and measure the isotopic ratios within a short time after separation to ensure there is no isobaric interference at mass 241. A standard chemical purification including redox cycle was applied on an aliquot of about 500 µg plutonium. The plutonium was absorbed on a anion exchange resin (Bio Rad AG1-X4 100-200 mesh) and eluted with 0.35M $HNO_3$. This solution containing the separated plutonium was then dried, taken up on 8M $HNO_3$ and separated a second time on an anion exchange column, eluting the plutonium again with 0.35M $HNO_3$.
The second separation was carried out so that measurement of the isotope ratios could be done within 48 hours. Under these conditions, the systematic errors introduced by the small amount of $^{241}$Am grown-in in the short time period, are negligible (<0.02%). This point was previously investigated in detail [1] and the possible presence of $^{241}$Am eliminated as source of interference in the half-life calculations.

### Sample loading

The samples were loaded in a glove-box onto filaments for mass spectrometric measurements. For the single-collector TIMS samples of 1 µg sample size were loaded and measured by peak jumping. On the multi-collector mass spectrometer, samples of 200 ng were loaded applying the total evaporation technique.

### Mass spectrometric measurements

For both mass spectrometers, a set of six or seven measurements of the isotope amount ratios concerned were performed within a period of 48 h after chemical separation and calibration of the spectrometer. One measurement corresponds to a single load of a sample into the mass spectrometer.
For the measurement by single Faraday on a MAT261 TIMS the method is essentially the one applied in 1997 [1]. For each filament, six groups of isotope amount ratios were measured over a period of time. The resulting isotope amount ratios per group were fitted as a function of time and the values interpolated 60 min after the start of the acquisition were used in all calculations. The internal standard deviation is a monitor for the stability of several measurement parameters. It is also an indicator of the presence of isobars. An external standard deviation was also calculated from the results of the measurement of the samples of a turret (six or seven).
The measurements on a MAT262 multi-collector TIMS were performed in static total evaporation mode. The ion beams of all isotope masses $^{238}$Pu, $^{239}$Pu, $^{240}$Pu, $^{241}$Pu and $^{242}$Pu are detected simultaneously throughout the entire measurement time for each filament. The filament temperature was adjusted in order to keep the ion current intensities at an appropriate level until the sample is completely evaporated and consumed for the data acquisition. The total evaporation method essentially removes mass fraction ('mass bias'). Effects from the isotope ratio measurements as described in detail in references [4, 5]. Nevertheless, isotopic reference materials from the IRMM-290 series were measured to verify the absence of any fractionation effects.

## Calculations and discussions

For each of the fifteen series of measurement a chemical separation was performed at a time $t_i$. Following the original design a ratio of ratios is constructed for each of the fifteen series of measurement, the isotopic ratios $n(^{241}\text{Pu})/n(^{240}\text{Pu})$ and $n(^{240}\text{Pu})/n(^{239}\text{Pu})$ are measured and the double ratio $[n(^{241}\text{Pu})/n(^{240}\text{Pu})]/[\, n(^{240}\text{Pu})/n(^{239}\text{Pu})]$ is calculated.
The double ratio decrease exponentially with time. A regression procedure is applied to the logarithm of the ratio-of-ratio measurements as a function of time. A model was setup in the GUMworkbench which allowed introducing the individual values for both variables with the accompanying uncertainties. In addition to this, the model includes a contribution from systemic error (type B variables).

- Non-linearity of the regression. In the GUM workbench, a term $\delta lin$ is added. This term expresses a possible effect in the measurement which contributes to the uncertainty. The value $u(\delta lin)$ of 0.04% is assigned after inspection of the absolute difference between individual observation and the calculated value as given in each case.
- Stability of the measurement system. A term $\delta stab$ added to quantify effects related to the difference in operating variables over the period of time such as mass spectrometers (Teledyne, MAT261, MAT262), methodology (TE, peak jumping), chemical preparations and operators. The value $u(\delta stab)$ of 0.03% is estimated by monitoring the variability on the $n(^{240}\text{Pu})/n(^{239}\text{Pu})$ ratio over the period of 30 years.
- Half-life of $^{239}$Pu and $^{240}$Pu following Table 2.

| Pu isotope | decay constant $\lambda v \pm \delta$ (in $10^{-5}$ $a^{-1}$) | half-life $T_{1/2} \pm \delta$ (in a) |
|---|---|---|
| 239 | 2.8746(13) | 24113(11) |
| 240 | 10.565(11) | 6561(7) |

Table 2: Decay constants and half-life for Pu isotopes [6]

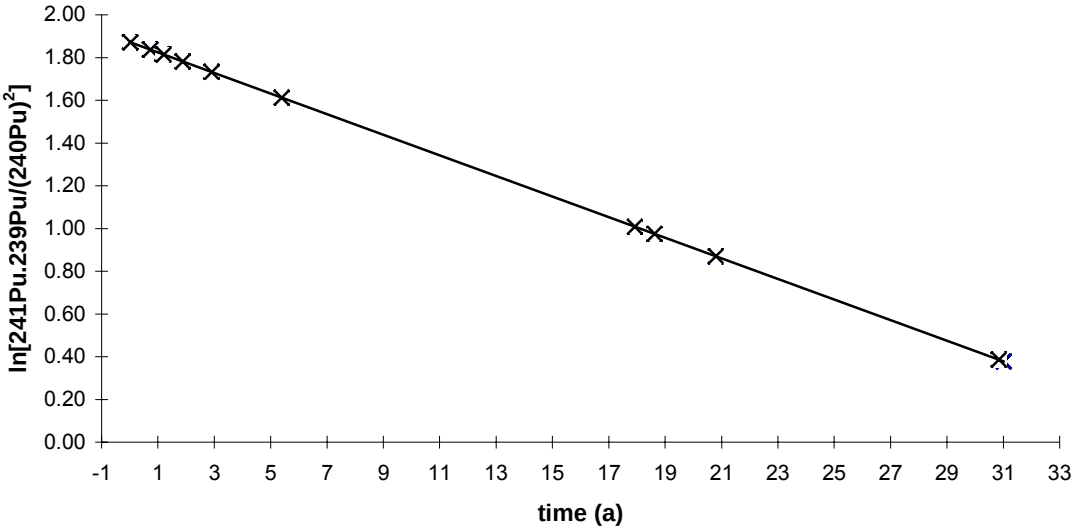


Figure 1: Regression on measurements

The residuals, the deviation between the regression line and each measurement point is shown in Figure 2.

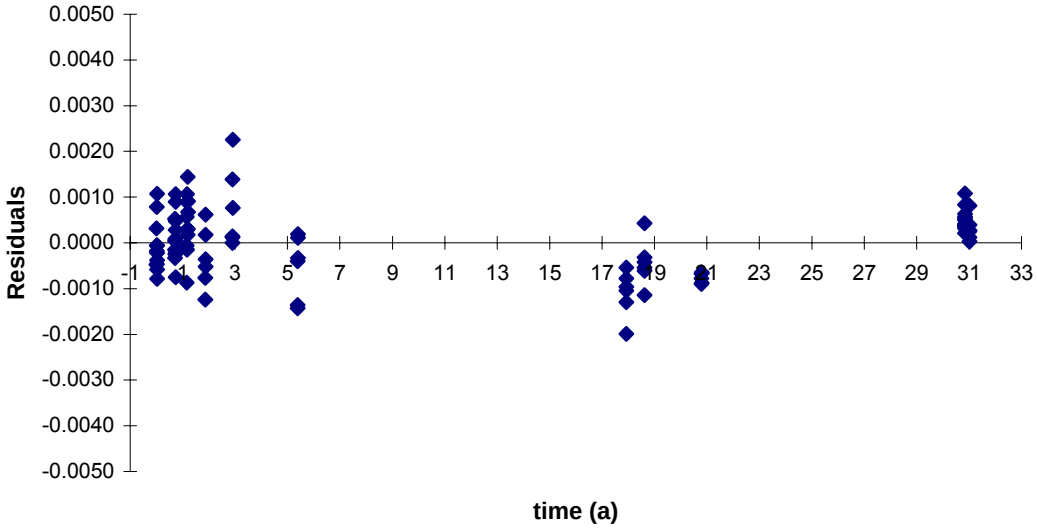


Figure 2: Residual, deviation from measurement values and line fit

The $n(^{240}Pu)/n(^{239}Pu)$ ratios were always measured within seconds before or after the $n(^{241}Pu)/n(^{240}Pu)$ ratios. Results in Figure 3 show:

- Reproducibility of "stable " decay corrected Pu isotopic ratio determinations
- Proof of freedom from isotopic contamination
- Proof of consistency of the measurement setup

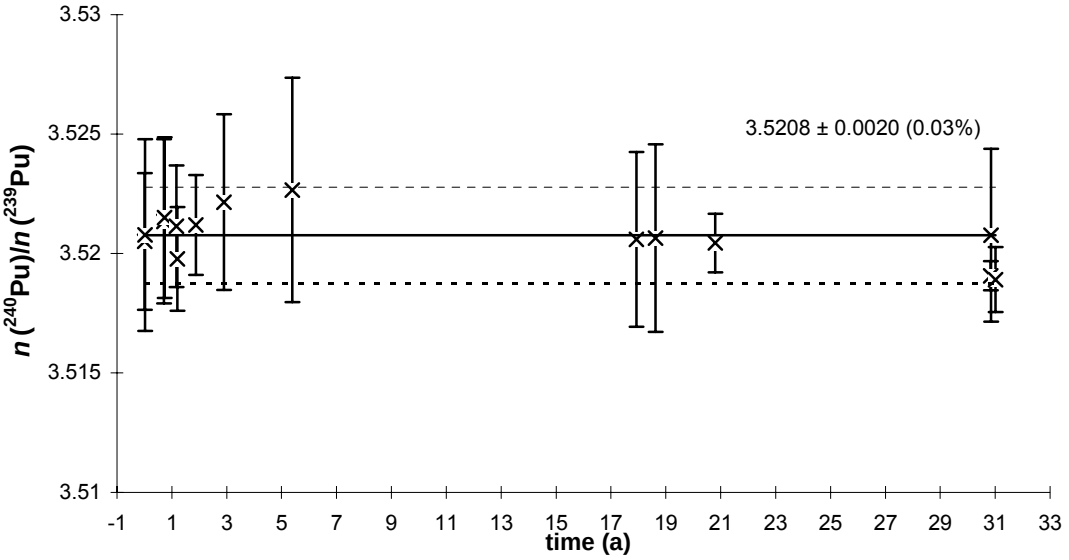


Figure 3: Reproducibility of $n(^{240}Pu)/n(^{239}Pu)$ measurements

Grouped measurements results and half-life determination obtained from this calculation are given in Figure 1. Calculations yield a value for the half life of 14.324 a, and a decay constant of 0.048391 $a^{-1}$.
During the course of the present measurements and calculations, the calculations from 1997 were found to inadvertently have included a logical mistake. A double correction for the decay of the isotopes, $^{240}Pu$ and $^{239}Pu$ was applied which lowered the value of the half-life of $^{241}Pu$.
An evaluation of components of uncertainty contributing to the combined uncertainty following GUM is shown in Table 3.

| Quantity | Value | Standard Uncertainty | |
|---|---|---|---|
| $t_i$ | 0.0077 - 31.011 | 0.0045 - 0.00013 | 3.1 % |
| $\ln[(241/240)/(239/(240)]_i$ | 1.872 -0.378 | 0.00069 | 58.3 % |
| $\lambda_{239}$ | $28.746 \cdot 10^{-6}$ | $0.013 \cdot 10^{-6}$ | 0.0 % |
| $\lambda_{240}$ | $105.65 \cdot 10^{-6}$ | $0.11 \cdot 10^{-6}$ | 0.0 % |
| $\delta lin$ | 0.0 | 0.0004 | 24.6 % |
| $\delta stab$ | 0.0 | 0.0003 | 13.8 % |

Table 3: Components of uncertainty

## Conclusions

A half-life for the β decay of $^{241}Pu$ of 14.324 a is being proposed. At present we are discussing suitable metrological methods of calculating the uncertainty given the number of data across 30 years.
The final value will be based on a correct metrological overview of the measured values leading to an uncertainty believed to be fit for purpose.

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# Session 30

# Material monitoring and Illicit trafficking

# Use of Imaging for Nuclear Material Control and Accountability

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***Abstract:***

The recent addition of imaging to the Nuclear Materials and Identification System (NMIS) using a small portable DT neutron generator with an embedded alpha detector to time and directionally tag neutrons from the DT reaction is discussed. The generator weighs ~35 lbs including power supplies ($5x10^7$ n/sec) and operates on 50 watts of power. Thus, the source can be easily moved to a variety of locations within an operational facility with minimum impact or it can be used at a fixed location for example to monitor receipts. Imaging with NMIS not only characterizes the detailed shape of a containerized object by transmission tomography but also determines the presence of fissile material by measuring the emitted radiation from induced fission. Imaging measurements with a time tagged Cf spontaneous fission source are also discussed. Previous studies have shown that this type of imaging has a variety of applications other than nuclear material control and accountability (NMC&A). These include nonproliferation applications such as verification of configuration of nuclear weapons/components shipped or received, warhead authentication behind an information barrier, and traceability of weapons components both fissile and non fissile in dismantlement and counter terrorism. This paper focuses on the use of imaging for NMC&A. Some of the NMC&A applications discussed includes: verifying inventory and receipts, making more accurate holdup measurements especially where thicknesses of materials affect gamma ray spectrometry, determining the shape of unknown configurations of fissile materials where the material type may be known but not the form, determining the oxidation of fissile metal in storage cans, fingerprinting the content of storage containers going into a storage facility, and determining unknown configurations for criticality safety.

**Keywords:** neutron, imaging, uranium metal, NMC&A, DT generator

## 1. Introduction

Tomographic and radiographic imaging capabilities have been added to the Nuclear Materials Identification System (NMIS) [1] at Oak Ridge National Laboratory (ORNL). The system uses transmission imaging with a time and directionally tagged portable DT generator [2 & 3] or a time tagged Cf spontaneous fission source [4]. Neutrons from the DT generator are more effective on penetrating of hydrogenous materials than Cf fission neutrons by a factor of ~5. However Cf is adequate for objects with low attenuation and where measurement time is not a factor. The generator has an additional advantage in that it can be turned off when not in use. These imaging capabilities supplement the system's existing time-correlation (coincidence) measurements to quantify fissile material and enhance the system's ability to identify fissile material in containers. This capability has a variety of application in NMC&A such as determination of the fissile mass holdup in pipes; especially in situations where heavy deposits produce self shielding that make gamma ray spectrometry determinations of hold up have large uncertainties. One such holdup measurement was satisfactorily performed in 1998 or the so called "hockey stick" deposit at the

K-29 building of the former gaseous diffusion plant at Oak Ridge [5]. The present NMIS with imaging can satisfy all DOE requirements for confirmation of receipt of weapons components at the Y-12 plant and its continued development will lead to a system that accurately measures the fissile mass in Y-12 receipts. Other than identification and confirmation of weapons components there a variety of other applications such as: determining the form of legacy materials such as highly enriched uranium (HEU) in storage cans where the form is not known, fingerprinting the configurations of HEU going into long term storage facilities, determining the fissile mass in plant components that process oxide and other materials without cleaning out the components, comparing the images of fissile materials in shipping containers where the detailed shape is available from the shipper, determining the amount of oxidized metal in storage cans for fissile metal, determining unknown configurations of fissile materials for criticality safety, and identifying appropriate standards for other NMC&A measurements on cans where internal configuration is not known, etc. The present status of this system is described in this paper and examples of imaging of objects with both a DT generator and a Cf source are presented with some comparisons of results for both sources.

## 2. Methodology

Figure 1 shows a plot of the time distribution of counts after the DT reaction. Three regions are of interest. In region 1, the gamma rays from induced fission and inelastic neutron scattering are detected first because they travel to the detectors with the speed of light and arrive before the transmitted neutrons. In region 2, the 14.1 MeV transmitted neutrons arrive at the detector. Region 2 also contains forward elastically scattered neutrons that have little change in direction and energy for HEU. Finally, in region 3 the neutrons and gamma rays from induced fission in the HEU are detected and arrive with an amplitude and time dependence characteristic of the HEU mass and multiplication. Region 2 is used for the imaging measurements, while region 3 is used to distinguish HEU from depleted uranium (DU).

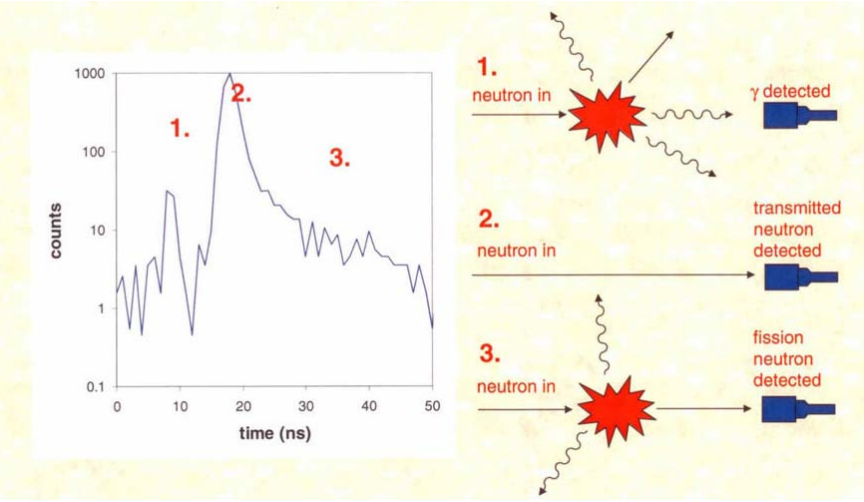


**Figure 1: Typical time distribution of counts in a plastic scintillator after the DT reaction in a transmission measurement through a highly enriched uranium metal**

Imaging with a DT generator consists of measuring the detector counts as a function of time after the DT reaction with the sample between the source and the detectors I(t) and the count rate as a function of time without the sample present $I_0(t)$ using the relationship $I(t) = I_0(t)exp(-\mu x)$, where $x$ is the sample thickness, and $\mu$ is the attenuation coefficient to obtain the value of $\mu x$, the attenuation. The values of $I$ and $I_0$ are obtained by integration of the transmission peak shown in region 2 of Figure 1 with an empirical correction to remove some scattering. This imaging was

performed as a function of height using a scanner that could rotate the eight small 1 × 1 × 6-in.-thick detectors 15° in the horizontal plane to simulate a larger number of detector positions.

The time distribution of counts for 4 plastic scintillators spaced ~100 cm in air from the source for a time tagged Cf spontaneous fission source is shown in Figure 2. The initial peak at the time of flight of gamma rays (~3 nanoseconds) from the source fission to the detector is the prompt gamma rays and is broadened by the time resolution of the detection systems for detecting spontaneous fission in the time tagged source and the event in the detector. The following peak (25-80 nanoseconds) is the neutron distribution whose time dispersion is determined by the energy distribution of prompt neutron emitted in spontaneous fission of Cf. Beyond 80 nsec, the distribution is mainly from neutron scattering from the floor. Separation of the gamma rays and neutrons allows both neutron and gamma ray imaging, with neutrons sensitive to light materials and gamma rays sensitive to heavy materials. For neutron imaging the area near the peak of the neutron distribution was used while for gamma rays the integral of the gamma peak was used.

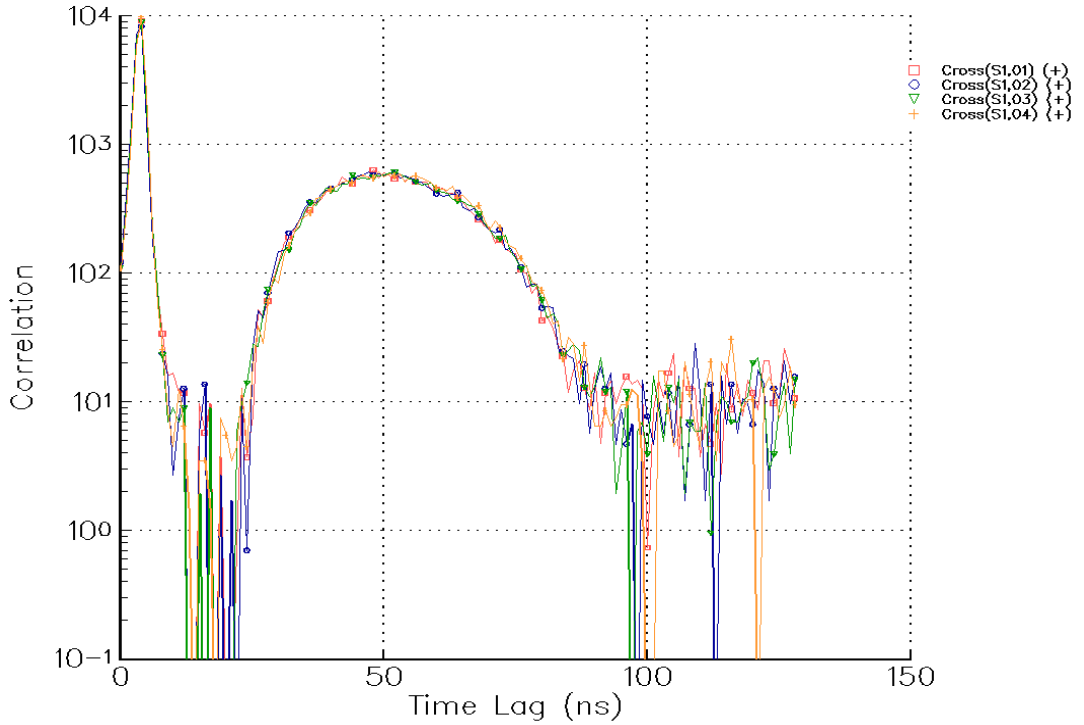


**Figure 2: Typical time distribution of counts from a detector after spontaneous fission of Cf. The initial peak is from gammas and the subsequent peak is from the neutron distribution.**

The source of preference is the DT generator with embedded alpha detector because of the penetrating capability of 14 MeV neutrons, shorter measurement time, and ability to turn it off when not in use.

## 3. Results for DU Metal Casting with DT Generator

A standard DU-metal annular Y-12 National Security Complex storage casting in a steel can was available for imaging measurements at ORNL. This casting has an outside diameter of 12.70 cm, an inside diameter of 8.89 cm, and a height of approximately 14 cm, with a density of approximately 18.8 g/cm$^3$. The casting was inside a 0.05-cm.-thick sealed steel can with an outside dimension of 15.2 cm and a height of 22.9 cm. A photograph of the imaging

measurement for the casting on a rotational platform is shown in Figure 3, with the DT generator on the left and the eight 1 × 1 × 6-in.-thick plastic scintillation detectors on the right.

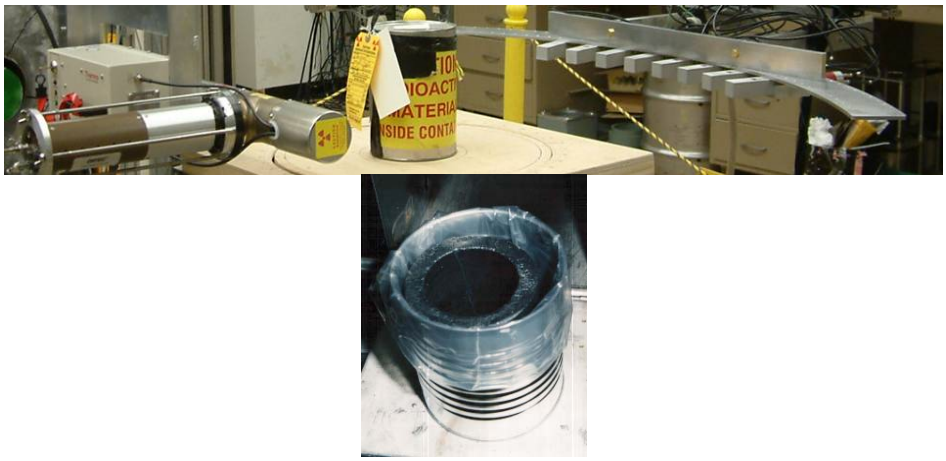

**Figure 3: Source-uranium metal-casting-detector configuration for imaging**

Initially, a vertical scan was performed over the full height of the casting, and the results are shown in Figure 4. A full vertical scan at a single projection angle provides an overall picture of the object to be imaged and allows the user to choose a particular height to image further. While the vertical scan shown in Figure 4 was obtained with high resolution at each level, these vertical scans can often be taken much quicker with fewer pixels and shorter exposures without affecting the ability to locate an area of interest. Vertical scans can be interpreted in the same fashion as x-ray images. The DU casting on top of the rotational platform can clearly be seen in the image. Next, two high-resolution scans were made at one height through the casting. These projections were captured at two rotational positions 90° apart. For both positions of the rotational platform, the 1-in.-square detectors were shifted 1/3-in. for successive measurements until the imaging arc was fully covered. Measuring with finer detail than the 1-in. detector face yields some gain in image resolution, depending in part on the image reconstruction method used.

Figure 5 shows the attenuation values for each of these positions in both projections (0° and 270° rotational positions). For a cylindrically symmetric casting, this 90° rotation will confirm symmetry and determine the position of the casting with respect to the center of rotation. For objects that are not cylindrically symmetrical about the axis of rotation, more projections are required to obtain the shape of the contents of the can. The NMIS analysis software includes a model-based fitting algorithm to determine geometric dimensions and attenuation coefficient values. This algorithm begins with the user entering an initial geometry of basic shapes from data such as that shown in Figure 4 and selecting key parameters (positions, sizes, and attenuation coefficients), allowed to vary during the fit. The fit uses an unconstrained nonlinear optimization algorithm in conjunction with a simulation tool based on ray tracing to find a fit of the measured projections to those generated by simulating the geometry. The results of fitting the attenuation projections at 0° and 90° rotational positions are shown in Figure 4, and the dimensions obtained are given in Table 1.

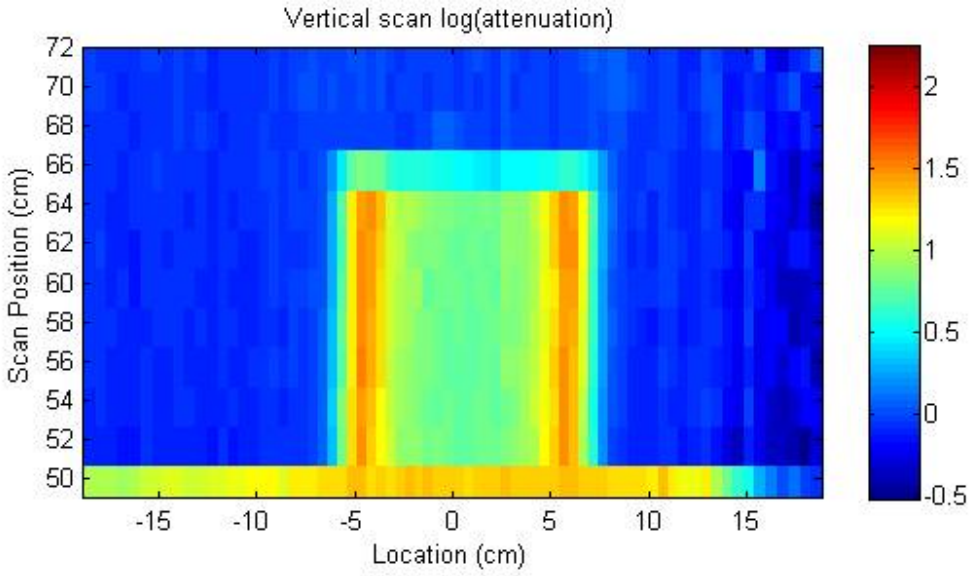


**Figure 4: Full scan of DU casting: lag of attenuation versus lateral location and height above the floor**

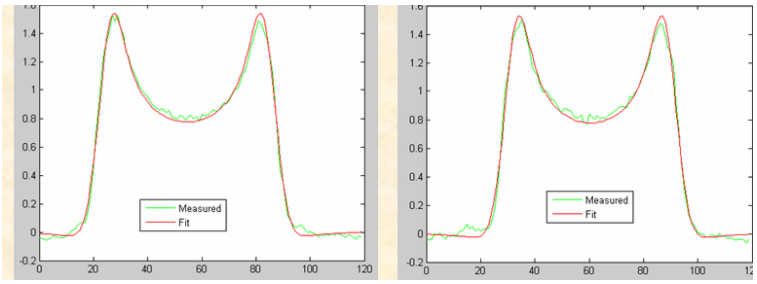


**Figure 5: Attenuation projections at 0° and 90° rotational positions and the results of fitting the data**

| Parameter | Initial guess | Final fit | Actual dimensions |
|---|---|---|---|
| Inner diameter (cm) | 10.00 | 8.56 | 8.89 |
| Outer diameter (cm) | 11.00 | 12.93 | 12.81 |
| Attenuation coefficient (2/cm or $cm^{-1}$) | 0.280 | 0.202 | 0.280 |
| X offset (cm) | 0.00 | -0.418 | Unknown |
| Y offset (cm) | 0.00 | 0.570 | Unknown |

**Table 1: Dimensions in centimeters from fit of casting data set containing two projections 90° apart**

The fitted dimensions are close to the actual. These measurements were performed with 1- by 1-in. pixels (detectors). Reduced pixel size would improve the accuracy of the dimensions. The fitted attenuation coefficient (0.202 $cm^{-1}$) is lower than the actual (0.28 $cm^{-1}$) because neutrons scattered at small angles from uranium still arrive at the detector and at nearly the same time as the directly transmitted neutrons. With proper correction for scattering, the values of the attenuation coefficients can be fixed for known materials, and more accurate dimensions can be obtained. Based on the X offset in Table 1, the axis of the casting was approximately 0.42 cm farther from the detector array than the axis of rotation. A second set of experiments was

performed on the DU casting by placing shielding around the casting. Two different shielding configurations were measured: (1) a 1-in.-thick lead box surrounding the steel can and (2) a 1-in.-thick lead box plus 4-in.-thick lead plates and bricks added to the front and back as shown in Figure 6.

**Figure 6: DU casting within a 1-in.-thick lead box surrounding and 4-in. thickness of lead plates in front and back**

In an effort to determine whether the internal components still provide a useful signature, a second set of projections was obtained for just the shielding without the DU casting and used to normalize the projections ($I_0$). The measurement times for the bare casting and the casting with 2 in of lead were 2 minutes and that for the casting with 10 in of lead was 10 minutes. The images for the different configurations of lead shielding are shown in Figure 7. Based on these results, the casting is still distinguishable with 5-in.-thick lead shielding front and back. Thus, the transmission was measured through 10 in. of lead and the casting. This result illustrates that if the shielding or container material is known and available, a measurement of transmission, $I_o$, with the shielding or container in place can normalize out the container and shielding effects.

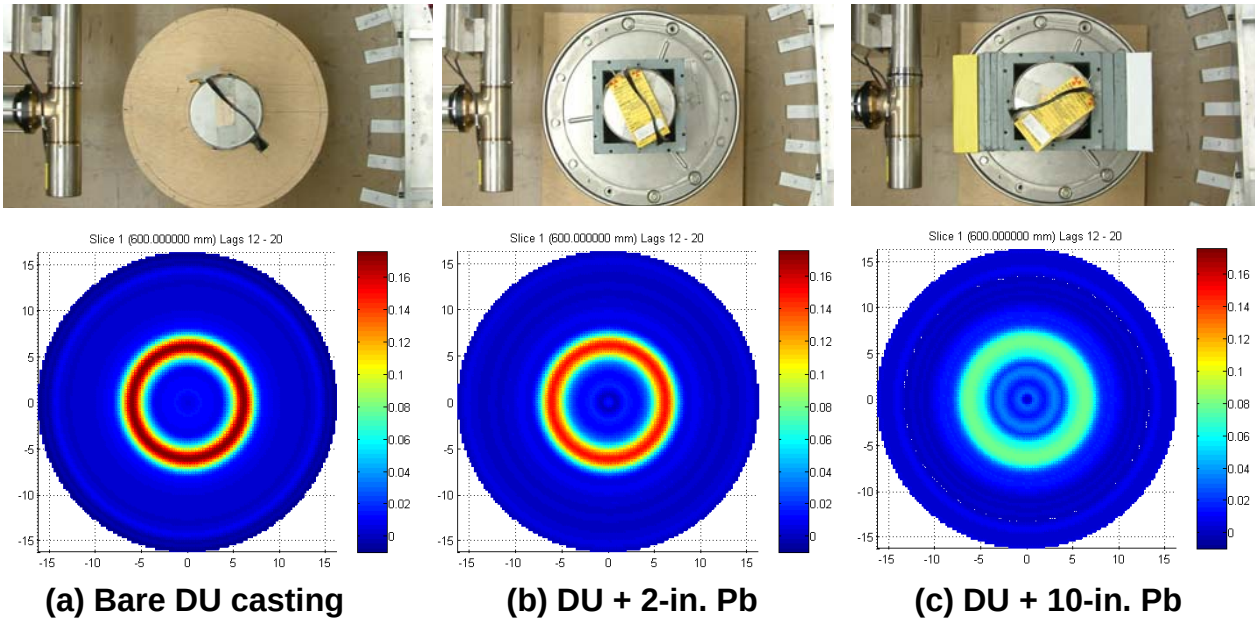


**Figure 7: Various shielded casting configurations and resulting images**

## 4. Measurement of a 3D Object with Cf

Imaging measurements were performed for a variety of objects that were not cylindrically symmetric. In this case the object needs to be rotated to measure the projection at various angles. One such object located is shown in Figure 8. The bottom of this object consists of an 8-in-square lead with a 6.5-in.-diam. central hole that is 6.75 in high. Spaced 0.5 in above that is a 0.5-in-thick, 7-in-square piece of Plexiglas. On the Plexiglas is a 3.5-in-high, 6.5-in-OD, 5.5 -in – ID lead part who's outside diameter increases to 7 in. for the top 0.75 in. Inside the top piece of lead and resting on the Plexiglas is a 3.5-in.-diam., 2-in.-high polyethylene cylinder adjacent to the inside surface of the lead. These features of the object are visible in Figure 8. This was a contrived test object used for training to test the ability to image an unknown 3D object. The object was then inserted into a drum and located as in Figure 9 between the Cf source (on the left) and 16 plastic scintillators (on the right) on a radial arm that could be rotated. The object was raised off the bottom of the drum by a low mass metal ring and plate.

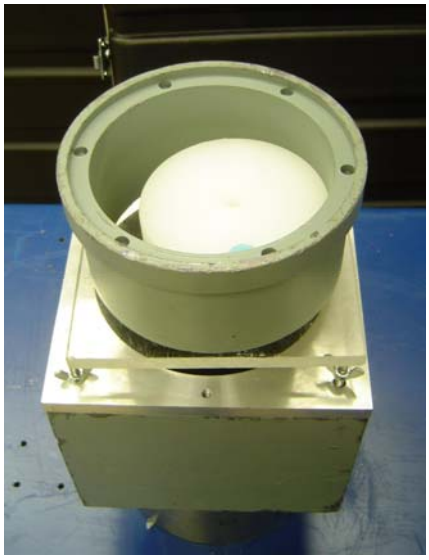

**Figure 8: 3D object of lead, aluminum, polyethylene, and Plexiglas**

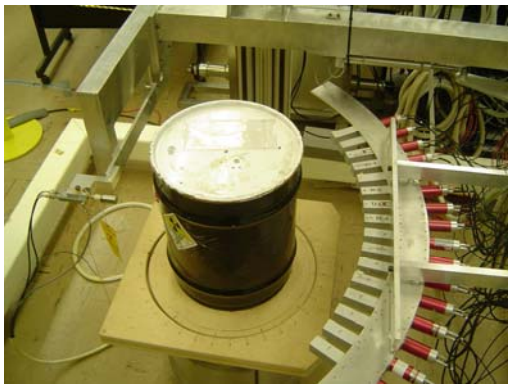

**Figure 9: 3D test object in a container on the rotating table between the Cf source on the left and the detectors on the right**

The initial measurements were a vertical scan of the object in the container with the arm supporting the detectors rotating laterally to simulate other radial detector positions. The results of these measurements are shown in Figures 10 and 11 where two dimensional plots of the attenuation length are presented. The polyethylene and the Plexiglas are clearly visible in the neutron scan whereas they are not in the gamma scan. This is not the case when imaging with a DT generator which is sensitive to both light and heavy materials [2].

Based on these vertical scans, a detailed image was measured at 2 vertical locations through the middle of the lower lead and through the middle of the polyethylene. Since these objects were not cylindrically symmetric, the transmission was measured at 18 different projection angles. The reconstructed images are shown in Figures 12 and 13. Clearly, the configurations of the test object at the 2 heights are determined. Future work in imaging testing on 3D objects will employ an automated object turntable. The use of a DT generator with pixilated alpha detector for this type of measurement will facilitate much better imaging through removal of scattering result in shorter measurement time.

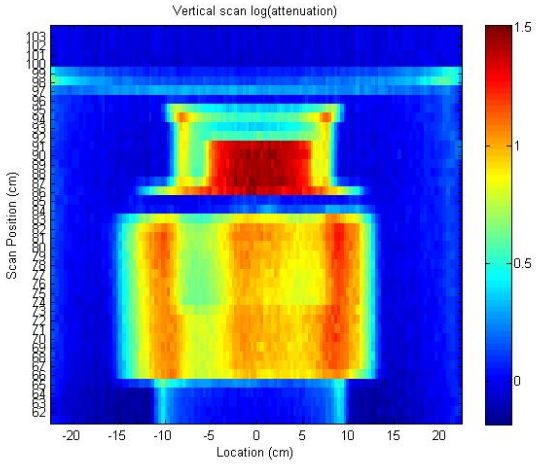


**Figure 10: Neutron radiograph of the test object from the vertical scan**

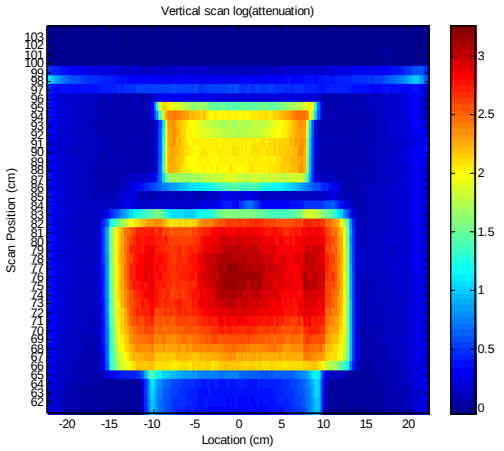


**Figure 11: Gamma radiograph of the test object from the vertical scan**

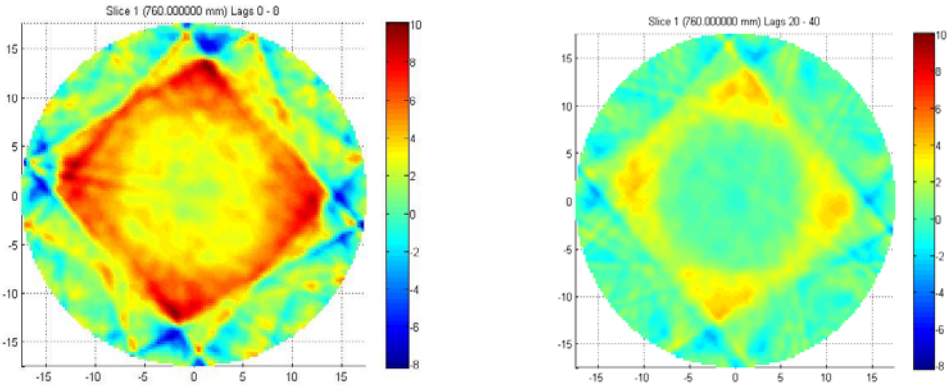


**Figure 12: Gamma and neutron images of the test object through the lower lead**

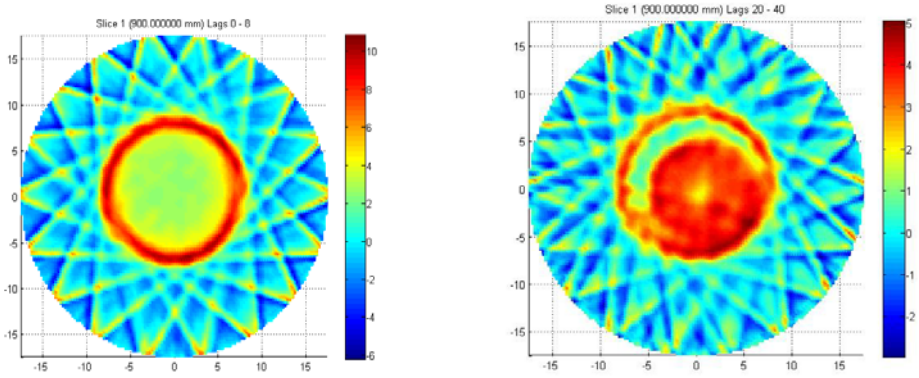


**Figure 13: Gamma and neutron images through the test object at the height of the polyethylene**

## 5. Applications

The imaging capability of NMIS can enhance NMC&A:

- Determination of the configuration of items in containers, and it is ideal for fissile material transfers between facilities. This capability is accomplished by providing the detailed configuration of the fissile and non-fissile components of an item under scrutiny. The images can be compared to drawings provided by the shipper avoiding repackaging delays to verify the contents of a shipment that may not be immediately used. Induced fission emitted radiation can be evaluated to distinguish fissile from non fissile materials.
- Periodic inventory confirmation.
- Determination of the form of legacy materials such as HEU in storage cans where the form is not known.
- Determination of unknown configurations of fissile materials for criticality safety and identification or to determine appropriate standards for other NMC&A measurements.
- Determination of the amount of oxidation of fissile metal in cans.
- Providing more accurate hold up measurements in pipes and process applications where the material and process vessel shape are known and process vessels can not be emptied. In some cases the density distribution in the container can be measured accurately so that the mass can be inferred to a few percent.

## 6. Conclusions

The imaging capability of NMIS using a DT generator with an embedded alpha detector can enhance NMC&A for a wide variety of applications addressing such problems as fissile transfers, fissile inventory, unknown fissile configurations, oxidation of fissile metal, and hold up in pipes and process vessels. This type of imaging has a variety of other uses such as verification of the configuration of nuclear weapons/components shipped or received, warhead authentication behind an information barrier, traceability of both fissile and non-fissile weapons components/parts during dismantlement, and counter terrorism.

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# An Inverse Transport Approach to Radiation Source Location for Border Security

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***Abstract:***

*Radiation portal monitors are being deployed at border crossings throughout the world to help prevent against the smuggling of nuclear and radiological materials. Many of these borders have several lanes for vehicles, each equipped with a portal monitor. With the current technology, if the detectors are alarmed, border guards must stop traffic and search for the source. In some cases, it can take hours to get through a busy border crossing. If radiation detection equipment adds a mere twenty seconds per car, this wait can increase by more than an hour. Another problem with these systems derives from the fact that one source can set off detectors in multiple lanes. If the source is being shielded by a vehicle in its lane, it may set off detectors in adjacent lanes but not its own.*

*The purpose of this research is to develop an algorithm for identifying the location of a radioactive source using a distributed array of detectors. To locate the source, some knowledge about the vehicles is needed. When a detector is alarmed, cameras installed in each lane will take a picture of the vehicles and a computer algorithm will build a cross section model of the traffic. The cross section model will be used in neutron and radiation transport calculations to determine the position of the radioactive material. There has been a lot of work done using inverse transport calculations to determine the material properties of an object, and this work uses some of the same techniques for source location. Forward transport calculations using a step-difference approximation are used to define an error functional describing the difference between the actual and calculated detector readings given an estimated source location. Adjoint transport calculations making use of a steepest descent method are used to minimize that error functional and thus identify the source location.*

**Keywords:** border; smuggling; inverse; transport; adjoint

## 1. Introduction

The United States Department of Homeland Security (DHS) has been deploying radiation detection equipment to the nation's border sites. One of the goals of this program is to screen all of the traffic coming into the U.S. without causing delays. The U.S. has over 380 ports-of-entry, and each day the DHS processes approximately 365,000 vehicles and over 1.1 million people arriving in the country. At the San Ysidro, California land border crossing it was estimated that prior to the installment of portal monitors it would take about 2.5 hours to get through the border at peak times. If the radiation detection equipment adds a mere 20 seconds for each vehicle this wait increases to 3.5 to 4 hours[1].

Many of these ports-of-entry have several lanes for vehicles, each equipped with a portal monitor. If a vehicle contains radioactive material, it may set off one or more of the detectors. Once detectors are alarmed Customs and Border Protection agents must stop traffic, locate the source, and identify the nature of the material. Not only is this inspection process time-consuming, there is also the threat that a source can set off detectors in adjacent lanes and not set off detectors in its own lane. If this happens, the source may get through the border. This underlines the need for a faster and more accurate way to detect radioactive sources at border crossings. The objective of this research is to

develop a method for identifying the location of a radioactive source using a distributed array of detectors.

For this work, detector measurements and information about the geometry and material properties of the vehicles will be used to triangulate the position of the source. If a source is detected, cameras installed at the border gates will take a picture of each lane of traffic and image recognition software will be used to estimate the types of vehicles (sedan, truck, van, etc.) in each position. Preconstructed cross section models for each type of vehicle will be used to make a cross section model of the entire system. This model will be used in an inverse calculation to determine the expected location of the source. That source location is used with the cross section model in a forward calculation to determine the expected detector signals. If the actual and expected detector measurements are equal, then the source position has been identified. If they are not equal, then the process is iterated. The solution method is outlined in Fig. 1.

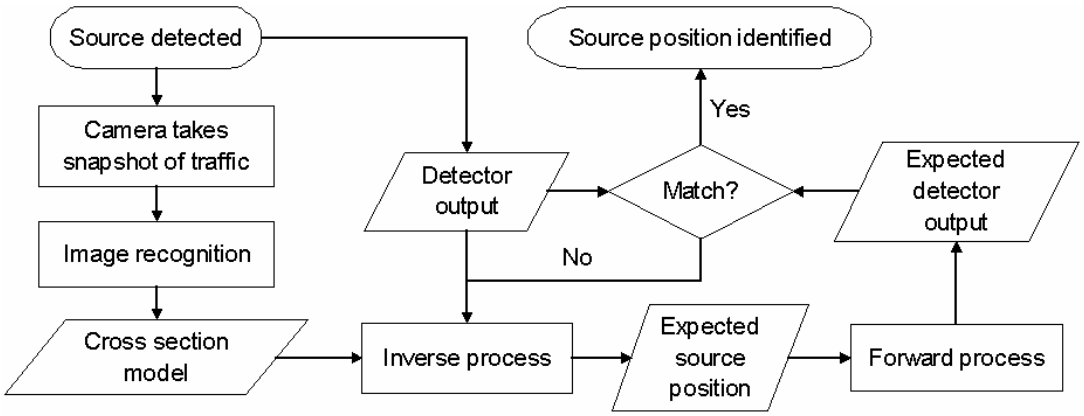


**Fig. 1:** General solution method for locating radiation sources.

All of the steps on the left column of Fig. 1 are outside the scope of this paper. The following sections assume a known cross section model and detector output. The discussion includes the work that has been done on the forward and inverse processes.

## 2. Theory

Inverse theory is the set of methods used to extract useful inferences about the world from physical measurements[2]. Extensive research has been done on inverse transport problems for determining material properties of a system[3-6]. This work uses some of the same principles for determining the location of a source.

The forward process solves the neutron transport equation with inhomogeneous source Q given by

$$\underline{\Omega}\cdot\underline{\nabla}\Psi\left(r,\underline{\Omega}\right)+\sigma_t\left(\underline{r}\right)\Psi\left(r,\underline{\Omega}\right)-\frac{\sigma_s\left(\underline{r}\right)}{4\pi}\int_{4\pi}d\Omega'\,\Psi\left(r,\underline{\Omega}'\right)=Q\left(\underline{r},\underline{\Omega}\right). \tag{1}$$

For brevity this is written as

$$L\Psi=Q\,, \tag{2}$$

where L is the transport operator defined by Eq. (1). The adjoint transport equation is

$$-\underline{\Omega}\cdot\underline{\nabla}\Psi^*\left(r,\underline{\Omega}\right)+\sigma_t\left(\underline{r}\right)\Psi^*\left(r,\underline{\Omega}\right)-\frac{\sigma_s\left(\underline{r}\right)}{4\pi}\int_{4\pi}d\Omega'\,\Psi^*\left(r,\underline{\Omega}'\right)=Q^*\left(\underline{r},\underline{\Omega}\right), \tag{3}$$

which can be shortened to

$$L^*\Psi^* = Q^* . \tag{4}$$

The adjoint source is defined as the difference between the calculated and actual detector response. It is given by

$$Q^* = \sigma_d \left( \Psi_{\text{det}} - \Psi \right), \tag{5}$$

where the cross section of the detectors is $\sigma_d \left( \underline{r} \right) = \sum_{n=1}^{N} R_n \delta \left( \underline{r} - \underline{r}_n \right)$, for N detectors. Assuming vacuum boundary conditions, the duality principle reads[7]

$$\int_V d^3 r \int_{4\pi} d\Omega \Psi^* L \Psi = \int_V d^3 r \int_{4\pi} d\Omega \Psi L^* \Psi^* . \tag{6}$$

If the system parameters, such as cross sections or source position, are perturbed, then there will be a change in the flux. Using the chain rule, the change in the transport equation is given by

$$\delta L \Psi + L \delta \Psi = \delta Q \text{ , or}$$

$$L \delta \Psi = \delta Q - \delta L \Psi . \tag{7}$$

The error in the detector response calculated with the forward equation with respect to the actual detector response can be described with a chi-squared error functional:

$$\mathrm{E} = \frac{1}{2} \int_V d^3 r \int_{4\pi} d\Omega \sigma_d \left( \Psi_{\text{det}} - \Psi \right)^2 . \tag{8}$$

The steepest descent method can be used to minimize the error functional[8]. To use this method, the gradient of the error functional with respect to the quantities of interest is needed. For example, the functional gradient (or Fréchet derivative) of the chi-squared error with respect to the x-position is given by

$$\nabla_x \mathrm{E} = \int_V d^3 r \int_{4\pi} d\Omega \frac{\partial \Psi}{\partial x} \sigma_d \left( \Psi_{\text{det}} - \Psi \right). \tag{9}$$

Using the Eq.'s (4) and (5) to replace the adjoint source and the principle of duality given by Eq. (6), this can be rewritten as

$$\nabla_x \mathrm{E} = \int_V dV \int_{4\pi} d\Omega \Psi^* L \frac{\partial \Psi}{\partial x} . \tag{10}$$

Replace the $L \frac{\partial \Psi}{\partial x}$ term using Eq. (7):

$$\nabla_x \mathrm{E} = \int_V dV \int_{4\pi} d\Omega \Psi^* \left( \frac{\partial Q}{\partial x} - \Psi \frac{\partial L}{\partial x} \right). \tag{11}$$

For the source location problem we assume that all of the cross sections are known, so $\partial L/\partial x = 0$. Also, the source is assumed to be a point source with strength q. Showing the source explicitly for a two-dimensional problem with x–y geometry, Eq. (11) becomes

$$\nabla_x \mathrm{E} = \int_V dV \int_{4\pi} d\Omega \Psi^* \frac{\partial}{\partial x}\left[ q\delta\left(x - x_0\right)\delta\left(y - y_0\right)\right], \tag{12}$$

where $\left(x_0, y_0\right)$ is the source position. To evaluate this integral, integrate by parts using

$$\int_{x_0-\varepsilon}^{x_0+\varepsilon} dx \frac{d}{dx}\left[\delta\left(x - x_0\right)\right] f\left(x\right) = -\left.\frac{df\left(x\right)}{dx}\right|_{x=x_0}. \tag{13}$$

Thus, the functional gradient of the chi-squared error with respect to the x-position is

$$\nabla_x \mathrm{E} = -q\left.\frac{\partial \Phi^*\left(x, y_0\right)}{\partial x}\right|_{x=x_0}, \tag{14}$$

where $\Phi^*$ is the adjoint scalar flux. The functional gradients of the chi-squared error with respect to the y-position and source strength are found similarly. They are

$$\nabla_y \mathrm{E} = -q\left.\frac{\partial \Phi^*\left(x_0, y\right)}{\partial y}\right|_{y=y_0}, \tag{15}$$

$$\nabla_q \mathrm{E} = \Phi^*\left(x_0, y_0\right). \tag{16}$$

In a steepest descent algorithm, each iteration updates the quantity of interest by some step size in the direction of steepest descent, which is the negative of the functional gradient. The step size can be calculated in several different ways. For the test problems in the following section, a 1D line search was used to determine the step size.

## 3. Test Problems

Test problems were run to look at two aspects of the algorithm: (1) the dependence of the solution on the initial guess for source position and (2) how error in the detector measurements affects the solution. Both test problems were run using the same two-dimensional system with x–y geometry. The neutron transport and adjoint equations were solved using a step-difference approximation and an $S_{12}$ quadrature set[9].

The test system was created to model a simplified-real-world scenario with two lanes of traffic and three detectors. Fig. 2 shows the system layout. The grey boxes are homogeneous–all with the same cross sections–and meant to represent vehicles. The dotted area between the “vehicles” is representative of air space. The detectors are the darker-grey circles. The configuration of the detectors is consistent with actual border crossings. The black square shows the source position. For both test problems the source strength was assumed known, so the algorithm identified source position only.

### 3.1. Initial Guess Test Problem

For this test the algorithm was run once for each cell in the two-dimensional grid. Each cell was tested as the initial guess for the source position. The objective of the test was to examine how the initial guess affects the solution. Fig. 3 shows that the algorithm estimated that the source is at five different

locations, depending on the initial guess. Outlines of the vehicles and detectors are included in Fig. 3 for reference.

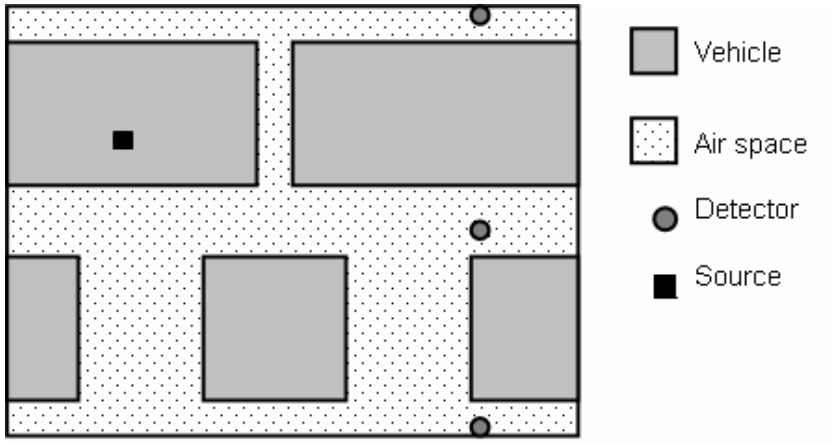


**Fig. 2:** Layout of the system for the test problems.

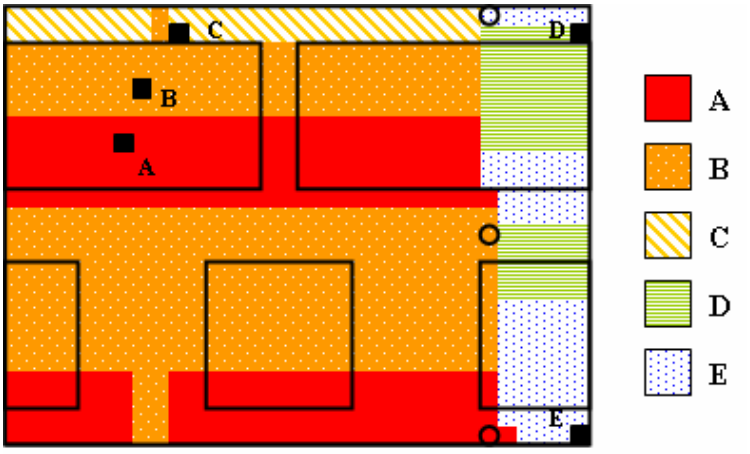


**Fig. 3:** Estimated source position based on the position of the initial guess.

The estimated source locations are marked in Fig. 3 with a black square and a corresponding letter. Point A is the actual source location. The key for Fig. 3 shows the shading that corresponds to each of the five estimated source locations. For example, all of the cells shaded in solid red converged to point A when those cells were used as the initial guess.

Fig. 3 shows that using most of the cells in the system as the initial guess resulted in either the correct source position being identified (point A) or a point inside the correct vehicle being identified (point B). Using the cells towards the top of the system and to the left of the detectors as the initial guess resulted in point C being identified as the source location in most cases. Point C is just outside of the vehicle that contains the source.

The reason that the solution converged to different positions has to do with the method for minimizing the error functional. As the name implies, the steepest descent method pushes the guess along the path of steepest descent. If that path falls into a local minimum, the steepest descent method will not push the solution out of the local minimum. All of the points estimated as the source location are local minima, and point A is the global minimum.

Using cells to the right of the detectors as the initial guess resulted in the estimated source location being at the right boundary of the system (points D and E). Points D and E are clearly not even close

to the actual source position; however, this result is not surprising. Points D and E are a result of the steepest descent method combined with the configuration of the detectors. Starting with cells to the right of the detectors, the steepest descent algorithm pushed the initial guess on a path away from the detectors. Because the detectors are in line with each other vertically, the difference between the calculated and actual detector readings would have gotten larger (maximizing the error functional) if the source was pushed to the left. If the detectors had been staggered vertically or if there were more detectors towards the left side of the system, the steepest descent method may have pushed the solution in the correct direction.

### 3.2. Detector Error Test Problem

The second test problem examines the affect of error in the detector measurements on the solution. The same two-dimensional system was used for this test as was used in the first test. The first test assumed that the detector measurements were perfect. Here, the same initial guess was used for all of the tests, but the error in the detector measurements was varied from 0% to 100% in increments of 5%. The initial guess was a cell located near the bottom detector, and it converged to the actual source position when there was 0% error in the detector measurements.

The error in the detector measurements was sampled from a Gaussian distribution, and it was assumed that all three detector measurements were increased by the same average percent error. The distance between the actual source position and the calculated source position divided by the length (in the x-direction) of the system as a function of the error in the detectors is plotted in Fig. 4.

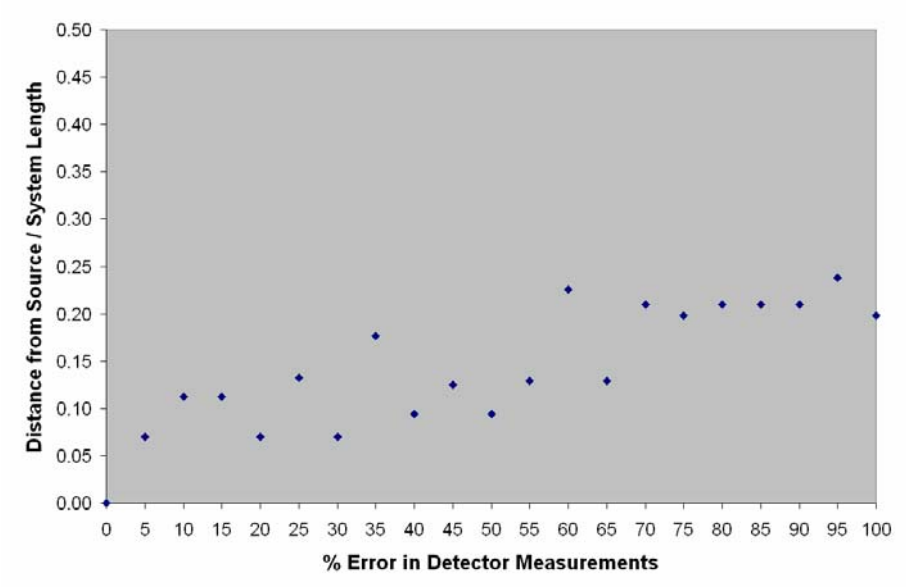


**Fig. 4:** Distance of the estimated source position from the actual source position divided by the system length as a function of error in the detector measurements.

The plot shows that increasing the error in the detectors caused the predicted source position to move away from the actual source position. For example, 10% error in the detector measurements caused the predicted source position to be a distance approximately 12% of the length of the system away from the actual source position. When the detector error was increased to 80%, the predicted source position was a distance approximately 20% of the system length away from the actual source position.

It is difficult to tell how these results will scale in a larger, more realistic problem with more heterogeneities and detectors that contain error in differing percentages without further testing. The conclusion that can be drawn from this test is that error in the detectors may be a significant factor in how well the algorithm predicts the source location and will need to be considered as this research progresses.

## 4. Conclusion

An algorithm for identifying the location of a radioactive source using a distributed array of detectors has been developed using forward and adjoint transport equation solutions. It uses the steepest descent method to minimize an error functional that describes the difference between the actual and calculated detector response. Results have shown that the solution is dependent on the initial guess used in the steepest descent algorithm. Also, error in the detector measurements may be an important factor in locating the source position. More study is needed to determine the extent to which detector error affects the solution.

Future work will include improvement of the source location algorithm within the framework of detector error. Other minimization techniques, such as using the conjugate gradient method instead of the steepest descent method, will be tested. Additionally, radiation transport will be included because many of the portal monitors in use detect neutron and gamma radiation, and we will include a method for determining the source strength.

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# Characteristics of an Integrated Germanium Detector Based Gamma-Ray Spectrometer for Monitoring Systems

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***Abstract:***

*An integrated gamma ray spectrometer, incorporating a germanium detector with integral mechanical cooling, digital signal processing electronics, MCA, and communications has been developed. It is intended as a modular subsystem for wide application, including use in stationary or mobile systems for the detection of radioactive materials. The requirements for this application are good low and medium energy detection efficiency and excellent spectral peak resolution. In a situation where either the spectrometer or the material is moving, the Field of View (FOV) determines the time the material contributes to the spectrum. The absolute efficiency and background determine the minimum detectable or identifiable quantity for the material in the FOV. To characterize the expected performance, the absolute efficiency was measured for a single unit for energies from 59 keV to 1.8 MeV. The horizontal FOV was measured for several source positions in the pedestrian portal configuration. The vertical FOV, important to determine the detector spacing, was also measured. The background was measured in a typical pedestrian portal situation. The measurements presented show this unit can be applied to a wide variety of monitoring situations.*

**Keywords:** radioisotope; integrated systems; germanium detectors; HPGe; illicit trafficking; monitoring

## 1. Introduction

As a consequence of increasing efforts to prevent the illicit trafficking of nuclear materials, particularly across national borders, there is a growing requirement for monitors of all types capable of detection of radioactive materials. The precise form of the monitoring system depends on the Concept of Operations or “CONOPS” at the crossing or facility. Consequently, because of the increased level of monitoring, it is of growing importance that these monitors be resistant to all forms of incorrect result: false negatives, false positives and false alarms (“no signal” alarms) are all highly undesirable.

The requirement to avoid false positives and negatives implies that the system must provide highly reliable nuclide identification. The majority of gamma-ray emissions from Special Nuclear Materials (SNM) are in the 100 to 600 keV range, implying that the detection efficiency in this energy range will depend mainly on the surface area of the detector. In germanium detectors, for example, a depth of 46 cm absorbs 90% of all 400 keV gamma rays incident on the face of the detector. Thus, surface area is more important than depth. (For example, see [1])

Large area, high purity germanium (HPGe) detectors meet all of these needs, but have historically had one major limitation, namely the requirement that they be cooled to cryogenic ($LN_2$) temperatures.

Recent developments in high reliability cryocoolers have resulted in the increased use of mechanical cooling for HPGe. Certain types of Stirling-cycle mechanical coolers typically have a design life on the order of 50,000 hrs, or more, of continuous operation, thereby greatly reducing the need for service intervention. Advances in spectroscopic signal processing using Digital Signal Processing (DSP) techniques have improved the spectrum quality both in terms of resolution improvements and stability with respect to temperature change and other causes of long term change in the position of the gamma-ray peaks. DSP technology allows pulse-by-pulse corrections to be applied to the data stream to eliminate deleterious effects such as degradation of the resolution by periodic noise. [2]

A completely integrated, autonomous spectrometer has been constructed, comprising a large surface area HPGe detector, mechanical cooling, and DSP electronics. It may be used as a “building block” component for the simplified construction of portal monitors for pedestrians, packages, vehicles, cargo containers, and rail freight cars (goods wagons) as well as of vehicle and airborne mobile search systems. The modular spectrometer is referred to as the “Interchangeable Detector Module” (IDM).

The data shown below are for a pedestrian portal as defined in ANSI N42.38-2006 [3]. The pedestrian portal was selected because the small size is easier to test. The results show that a portal with relatively few large area HPGe detectors can meet the requirements of ANSI N42.38.

## 2. Equipment

The IDM consists of an 85 mm x 30 mm HPGe Detector, Stirling cooler, DSP MCA, high voltage supply, shielding against gamma rays from behind the front surface, and high speed USB communication. The instrument can be mounted in a standard electronics enclosure (rack mount), as shown in Fig.1. It uses standard, low-current mains power.

**Figure 1 Complete IDM**

The large diameter detector gives good low-energy efficiency. Standardized detector crystal dimensions mean that all IDMs will perform similarly so that efficiency recalibration is not necessary if an IDM is replaced in a system. Hence, the term “interchangeable”.

The uniform construction also allows computer modeling to be used to predict system performance. This means that customized solutions may be designed which will match the required CONOPS of a particular facility, for example, in terms of traffic flows and analysis speeds versus system cost.

The high cooling capacity, highly reliable, Stirling cooler will operate in the environmental conditions stipulated by ANSI N42.38-2006 without the need of external heaters or air conditioning units. The hardened cryostat is designed for long operational life and can be temperature cycled at any time, even from partial warm-up, eliminating the problems associated with loss of electrical power. If the power is turned off, it will automatically restart when the power is turned on.

The DSP MCA has 16k channels and can operate in histogram mode (standard PHA mode) or streaming list mode. Streaming list mode has no loss of data between spectrum captures and allows for off-line data manipulation to improve the detection ability. Raw or processed data can be sent to the controlling computer over the high speed USB connection. The IDM can operate and collect data without a PC, but the normal operating mode would use a PC for data display.

The field-of-view measurements were made using the commercial ORTEC GammaVision software, and Region-of-Interest analysis of the peaks from $^{133}$Ba, $^{57}$Co, and $^{60}$Co. The detection and identification was done using a method similar to that used in the ORTEC Detective Handheld Identifier. This method has been shown to perform well in the short data collection time expected in the portal monitoring application.

### 2.1. Experimental conditions

For the standard efficiency measurements, the IDMs were positioned on a table with the mixed gamma point source (gamma-ray energies from 59 keV to 1.8 MeV) at 25 cm from the front face of the endcap and centered on the endcap. While the portal monitor application is an identification application and does not typically produce activities, measuring the efficiency in this way (IEEE 325-2006) allows comparison with other HPGe detectors. The energy resolution (FWHM) was measured using the same spectra.

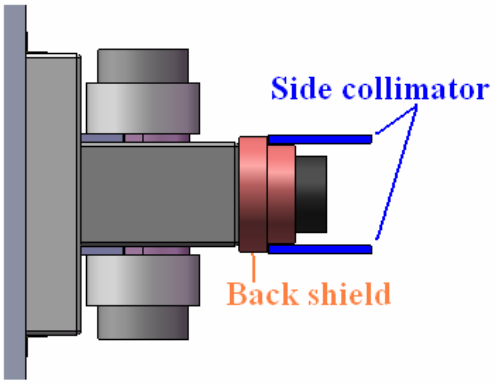


**Figure 2 Collimators on the IDM as seen from top**

The Field-of-View (FOV) measurements were done with the IDMs supported on adjustable supports with steel collimators to reduce the field of view. The steel collimators were positioned as shown in Fig.2. To cover the occupancy zone of the two-sided pedestrian portal, 4 IDMs (2 on each side) were positioned at vertical positions of 50 cm and 160 cm. The front faces of the detector endcap are 1 meter apart. The total assembly (without collimators on the right side) is shown in Fig. 3. The sources were moved using an automated positioner. The reproducibility of the position is about 1 mm. The FOV was measured using $^{133}$Ba, $^{57}$Co and $^{60}$Co.

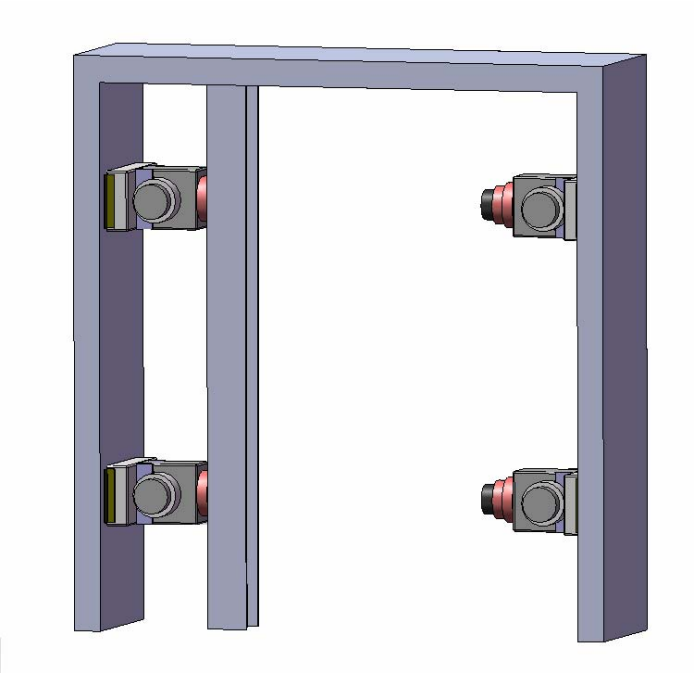

**Figure 3 Pedestrian Portal Configuration**

## 3. Results

### 3.1. Efficiency

A total of 8 IDMs were measured. The absolute efficiency for all IDMs is shown in Fig. 4. This shows the typical dependence on energy of p-type HPGe detectors and that the IDMs have efficiencies within 5% for energies above 100 keV. The efficiency at 59 keV depends on the crystal dead layer thickness which varies from detector to detector, giving this point a standard deviation of 15%. The standard deviation at 1332 keV is 1.7%

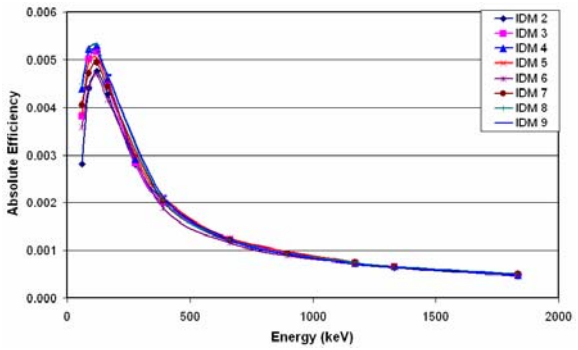


**Figure 4 Efficiency for Point Source at 25 cm for Several IDMs**

The average efficiency is shown in Fig. 5. The relative efficiency of the average detector is about 55%.

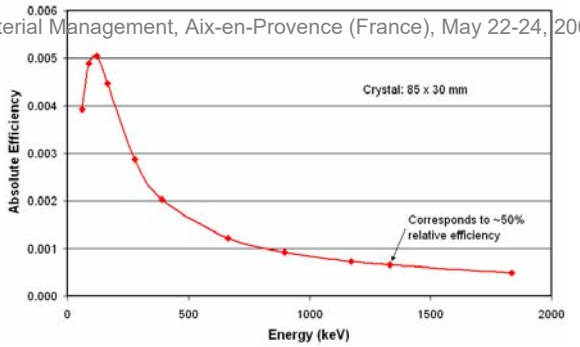


**Figure 5 Average Efficiency for IDMs**

## 3.2. Resolution

The resolution (FWHM) for all 8 IDMs is shown in Fig. 6. With one exception, the results are clustered together showing the detectors are similar enough to be treated equally by any analysis software. There is no explanation for the performance of IDM 5, that is, IDM 5 appears to be the same as the others in other aspects.

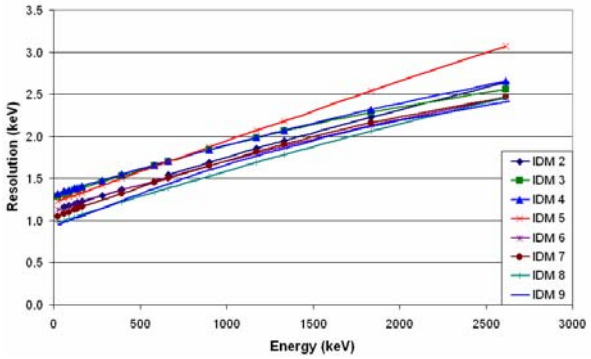


**Figure 6 Resolution of all 8 IDMs**

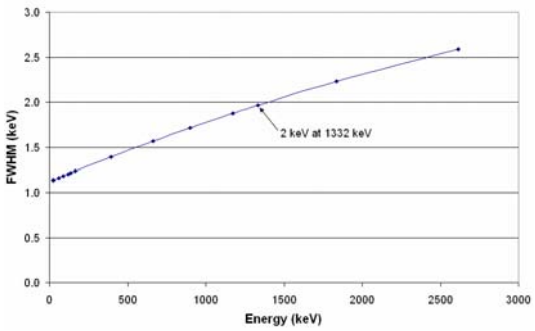


**Figure 7 Average Resolution of IDM**

## 3.3. Response along the vertical direction

The IDMs are uncollimated in the vertical direction. For the pedestrian portal, the detection zone is from 10 cm to 2 m above the floor. The response should be as uniform as possible for activity anywhere in this zone. Based on measurements of the detector response, the distance between the detectors was determined to have the best uniformity when the detectors were separated by 1.1 m. Figure 8 shows the relative response, at 383 keV, for 2 IDMs individually and the expected composite result. The source was moved vertically in a plane 1 m from the front face of the detector. The composite or sum represents the response of the two IDMs which are added to improve the detection ability.

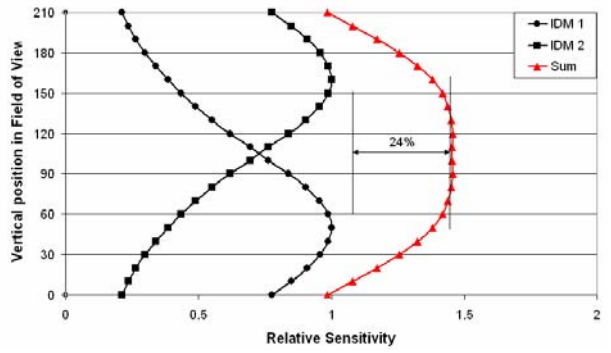


**Figure 8 Calculated Vertical Response for 2 IDMs**

## 3.4. Horizontal Field of View

The horizontal field of view (FOV) is reduced by collimators as shown above. The FOV is reduced to minimize the contribution from other pedestrians and the natural background. The collimator in this measurement was 12.5 mm steel plates positioned 19.7 cm apart with the detector endcap recessed 9 cm. The source was 1 m from the endcap. The width of 0.8 meters is the nominal horizontal length of the

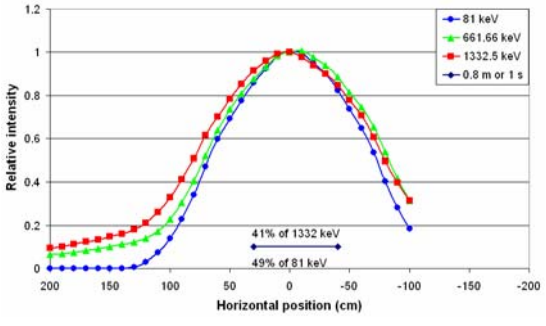


**Figure 9 Horizontal Field of View**

detection zone when using the ANSI N42.38 specification of a 1 second collection time and the pedestrian moving at 1.2 m/s. Data collected when the pedestrian was in this zone would represent about 49% at low energy and 41% at high energy of the possible data collected over a length of 4 meters. This is because the steel collimator is more effective at low energies. This indicates that more collimation could be used to reduce background and unwanted counts from the next person without reducing the data from the subject in the detection zone. The ability to collect data in the list mode (that is, a continuous time-stamped data log) enables the software to dynamically determine the time when the source is centered in the portal and select the time window for the best sensitivity.

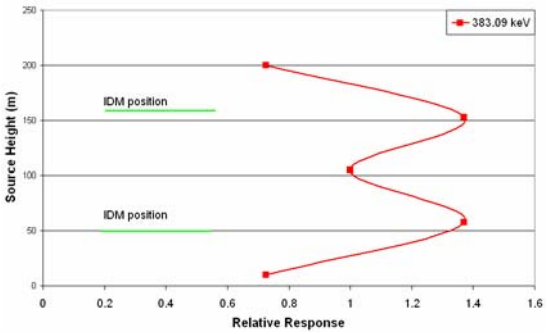


**Figure 10 Measured Vertical Response for 2 IDMs**

### 3.5. Response in the Detection Zone

The test positions for the source are in the middle of the horizontal direction of detection zone and at five vertical positions: bottom, 25% of height, middle, 75% of height, and top of the detection zone as required by the ANSI standard. For the pedestrian portal, this corresponds to a horizontal distance between the source and endcap of 50 cm and vertical distances from the floor of 10, 57.5, 105, 152.5 and 200 cm. The response at 383 keV, normalized to the midpoint (105 cm), for the sum of 4 IDMs is shown in Fig. 10 for these positions. This differs from the expected response shown in Fig. 8 mainly because of the difference in the source to detector distance (1 m above and 0.5 m here).

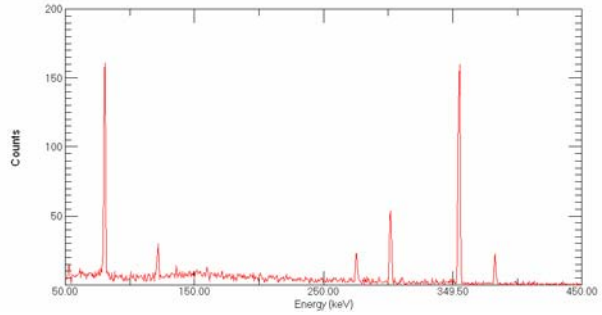


**Figure 11 Response for 1 s at Center of Detection Zone**

One measure of the minimum identifiable activity is the spectrum contents at the expected count time. The count time is fixed at 1 s in N42.38. Figure 11 shows the 1 second (real time) spectrum with the source stationary at the midpoint for the sum of 4 IDMs. Figure 12 shows the summed spectrum over a 1-second time window for 4 IDMs with the source moving through the portal. Both of these spectra were taken with 2.5 MBq $^{133}$Ba, 110 kBq $^{57}$Co and 728 kBq $^{60}$Co sources. The stationary spectrum has about 20% more total counts than the moving spectrum. In Fig. 12, the peak quality factor (Q) [4] for 122 keV peak is about 7.1. The Q threshold for the required false positive rate depends on the local background, but is generally set to 5. This gives the Minimum Identifiable Activity (MIA) of $^{57}$Co based on the 122 keV peak as less than 110 kBq for the collection time of 1 s moving at 1.2 m/s, which is lower than the detection activity of 185 kBq and the identification activity of 555 kBq given in N42.38.

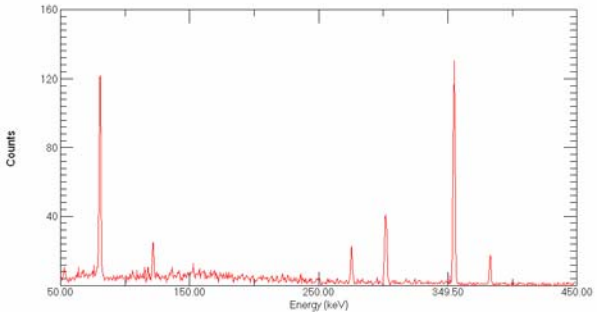


**Figure 12 Response for 1 s pass through**

## 4. Conclusions

The configuration of a pedestrian portal with 4 IDMs, each with a HPGe of 85 x 30 mm, has been tested for uniformity of response and sensitivity (efficiency). The detector placement was shown to have good uniformity in the detection zone as defined in ANSI N42.38. In addition, the number and size of detectors have necessary sensitivity to meet the detection requirements of the standard. Further work will show the response over the entire detection zone and determine the minimum identifiable activity.

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# Poster session

# Applicability of Transuranic Isotope Ratios for Origin Assessment

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## Introduction

Anthropogenic long-lived radionuclides (e.g. $^{237}$Np, Pu-isotopes and $^{241}$Am) have been released into the environment as a result of atmospheric weapon tests, nuclear power plant or satellite accidents[1]. Determination of plutonium and $^{241}$Am concentration is of high importance due to their radiotoxicity. Moreover, the isotopic composition is also of great interest for the evaluation of the possible origin of contamination, which can be exploited in environmental monitoring[2], nuclear safeguards and nuclear forensic studies[3]. Recently, study of deposited fallout plutonium used as a tracer is also a developing field in oceanographic studies and erosion investigations.

Improved sample preparation methods for the determination of plutonium and $^{241}$Am in environmental samples at ultratrace-level by inductively coupled plasma sector field mass spectrometry (ICP-SFMS) and alpha-spectrometry are presented. The methods were applied for the analysis of environmental samples collected from various contaminated and non-contaminated areas. The applicability of the methods and assessment of sources of contamination by its isotopic composition are discussed.

## Samples, Sample Preparation and Analysis

For the assessment of different sources of contamination various sample types (soil, sediment, biota) from different locations were analysed. In most cases commercially available reference materials were used due to their well-documented sampling circumstances, easy availability and wide variety of location. Furthermore, bioindicators, especially moss samples were measured originating from contaminated (collected during the Chernobyl expedition of Hungarian Nuclear Society in 2004) and non-contaminated areas (South-France, Hungary) for comparison. The sampling locations, together with sample types are shown in **Fig 1.** The sample preparation methods applied for the analysis that involves selective $CaF_2$ co-precipitation and three different extraction chromatographic separations are described in details elsewhere[4]. The mass spectrometric analysis of americium and plutonium was carried out using a double-focusing magnetic sector inductively coupled plasma mass spectrometer (ELEMENT, Thermo Electron Corp.,). Alpha sources were counted by a PIPS type alpha Si detector with a surface area of 450 mm$^2$ attached to an alpha spectrometer (Canberra Inc., USA). The detection limits by ICP-SFMS ($^{239}$Pu, $^{240}$Pu, $^{241}$Pu and $^{241}$Am) were in the 0.5-2 fg g$^{-1}$ range, with alpha spectrometry the detection limit of 0.08 mBq was achieved for $^{238}$Pu, $^{239+240}$Pu and $^{241}$Am.

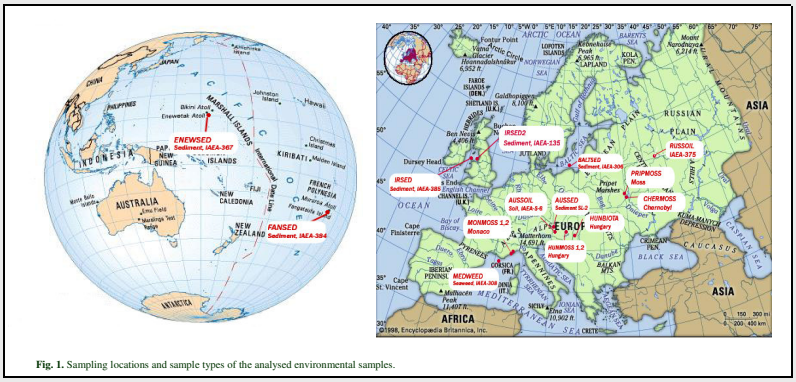


**Fig. 1.** Sampling locations and sample types of the analysed environmental samples.

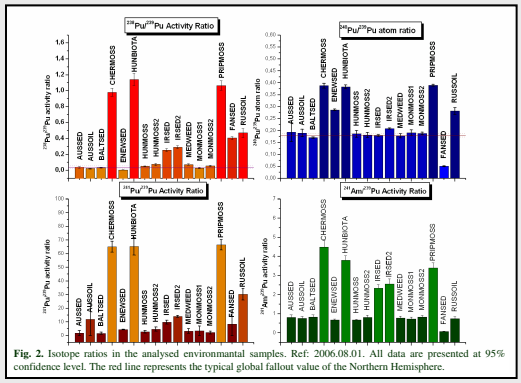


**Fig. 2.** Isotope ratios in the analysed environmantal samples. Ref: 2006.08.01. All data are presented at 95% confidence level. The red line represents the typical global fallout value of the Northern Hemisphere.

## Possible Origin of Transuranics

The possible origin of transuranic contamination can be assessed by the measured isotope ratios (**Fig. 2**.). In most measured low-level environmental samples collected in South-France and Hungary the Pu and Am ratios agree with those of global fallout of earlier atmospheric weapon tests. In some cases, in the Irish Sea sediments and Russian soil, the elevated level of higher mass plutonium isotopes and $^{241}$Am with respect to $^{239}$Pu indicates the contamination from Sellafield reprocessing plant and Chernobyl accident, respectively. Using the isotope ratios the contribution of reactor plutonium can be assessed assuming a two-component mixing model (**Table 1**.). The isotope ratios of the Hungarian biota sample (HUNBIOTA) collected after the Chernobyl accident in 1986 in Hungary agree with those of Chernobyl samples indicating that the accident is the major source of contamination. In samples collected close to previous weapon test sites in Oceania the plutonium isotope ratios show very high level of $^{239}$Pu suggesting the probable source of weapons-grade plutonium, though the elevated $^{240}$Pu/$^{239}$Pu ratio in Enewatok sample (ENEWSED) indicates very high neutron irradiation. Thus, in this case the sample possibly contaminated with plutonium from an H-bomb weapon test (fusion bomb) with extreme neutron flux, while Fangataufa sample (FANSED) contains plutonium that has similar isotopic composition to that of weapons-grade plutonium of a Pu-bomb (fission bomb).

## Release date Assessment of Transuranics

Assuming no chemical fractionation takes place between plutonium and americium, applying the decay laws one can calculate the elapsed time after formation of $^{241}$Pu as a function of $^{241}$Am/$^{241}$Pu activity ratio. The elapsed time between the formation of $^{241}$Pu and date of measurement can be calculated as follows:

$$t = \frac{1}{\lambda_1 - \lambda_2} \ln\left(\frac{R \cdot (\lambda_2 - \lambda_1) - \lambda_2}{R_0 \cdot (\lambda_2 - \lambda_1) - \lambda_2}\right)$$

where $\lambda_1$ and $\lambda_2$ are decay constants of $^{241}$Pu and $^{241}$Am, respectively, and $R_0$ and $R$ are the $^{241}$Am/$^{241}$Pu activity ratios at the time of formation and measurement, respectively. In case of environmental samples, however, it is more complicated to directly adapt the method, as following the release it must be ensured or controlled by proper sampling that there is no chemical fractionation between the $^{241}$Pu mother and $^{241}$Am daughter nuclide (**Table 2**.).

**Table 1** The contribution of global fallout and Chernobyl to the Pu content of the investigated samples.

| Sample name | Contribution from global fallout of weapons tests | Chernobyl contribution |
|---|---|---|
| AUSSED | > 98% | < 2% |
| AUSSOIL | > 98% | < 2% |
| BALTSED | > 98% | < 2% |
| CHERMOSS | < 5% | > 95% |
| HUNBIOTA | < 5% | > 95% |
| HUNMOSS | > 98% | < 2% |
| HUNMOSS2 | > 98% | < 2% |
| MEDWEED | > 98% | < 2% |
| MONMOSS1 | > 98% | < 2% |
| MONMOSS2 | > 98% | < 2% |
| PRIPMOSS | < 5% | > 95% |
| RUSSOIL | 56 ± 5% | 44 ± 4% |

**Table 2** The estimated release dates of some investigated samples.

| Sample | Calculated age | Estimated contamination date |
|---|---|---|
| CHERMOSS | 23 ± 2 | 1983 ± 2 |
| PRIPMOSS | 19 ± 2 | 1987 ± 2 |
| HUNBIOTA | 21 ± 2 | 1985 ± 2 |
| HUNMOSS | 47 ± 3 | 1959 ± 3 |
| MEDWEED | 43 ± 3 | 1963 ± 3 |
| IRSED | 45 ± 3 | 1961 ± 3 |
| FANSED | 36 ± 3 | 1970 ± 3 |

## Conclusions

- ✓ *Improved analytical procedures for determination of plutonium and $^{241}$Am concentration at ultratrace-level in environmental samples by ICP-SFMS and alpha-spectrometry were developed.*
- ✓ *Using the isotope ratios the possible origin of transuranics contamination can be assessed and their contributions can be evaluated.*
- ✓ *Using the $^{241}$Am/$^{241}$Pu activity ratio the release date of transuranics contamination can be assessed provided fractionaction effect is avoided*
- ✓ *The models can be applied for environmental monitoring and safeguards to control and detect former and recent transuranic contamination.*

## Acknowledgements

The authors would like to thank T. Bíró (Institute of Isotopes) for his helpful discussions. Part of this study was carried out within a Technical Cooperation with the IAEA at the Marine Anvironment Laboratory, Monaco and financially supported by the Hungarian Atomic Energy Authority (OAH-ANI-ABA-05/06).

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# Verification and disposal of PuBe neutron sources in Hungary

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***Abstract:***

*Verification of sealed PuBe neutron sources present in Hungary and other countries is a safeguards issue. The Pu content in these sources was not, however, declared upon delivery, and it had been basically unknown. The missing information is relevant for safeguards, nuclear safety, physical protection, illicit trafficking, and material management purposes. The facility and State inventories are still based on rough estimated values. For this reason, development of a task has been started in 2001, in order to solve the long-lasting problem. In the frame of this project different NDA methods have been developed for the characterization of these sources. By using these methods, the neutron output, isotopics, and Pu content of 76 PuBe sources (out of altogether ~200 sources in the country, most of them out of use) were determined. In this way, Pu amount to be accounted for reduced substantially. Results show that Pu mass of some sources has been overestimated even by an order of magnitude. The total Pu amount of all the 76 sources is 563 g instead of 2050 g, the sum of declared values.*

*After the measurements the sources were re-encapsulated in stainless steel holders to exclude Pu leakage into the environment (air), and were placed in special containers designed for transport and storage of neutron sources and nuclear materials. These containers, being accessible to IAEA and EURATOM inspections, are disposed in the preliminary store of nuclear materials out of use.*

*A new inventory of the Pu content of all the Pu-Be sources in the country is planned to be taken as well. The method is offered for routine IAEA use and also to other countries facing similar problems.*

**Keywords:** Non-destructive plutonium assay; PuBe neutron sources; neutron coincidence counting; gamma spectrometry.

## 1. Introduction

There are about 200 Pu-Be sources, most of them out of use, in Hungary. Whereas it is a safeguards and safety issue, their Pu content was not provided upon delivery. The supplier declared neutron output and activity only. However, the activity was obviously derived from the measured neutron output, by dividing it by a factor $1\times10^6$ for most of the sources, converting neutron output into activity in units of Ci. This corresponds to the lower limit of the (α,n) specific yield given as $(1–2)\times10^6$ n/s·Ci for $^{239}$Pu–Be sources in the literature [1], depending on the production technology. From the activity, a nominal Pu content was possible to be calculated, dividing it by the specific activity 2.3 GBq/g (0.0617 Ci/g) of $^{239}$Pu [2], while assuming a pure $^{239}$Pu content. The facility and State inventories are still based on these nominal values.

Since the specific activities of the various Pu isotopes differ substantially, the neutron yield depends very heavily on the actual isotopic composition. Our aim was therefore to determine it, and then to infer a more realistic Pu content of Pu–Be sources. The isotopic composition (isotopics) was determined by high resolution gamma-spectrometry (HRGS) which, on the other hand, is necessary anyway for identification of smuggled and seized, accidentally found, or not documented sources.

For determining neutron output (strength), a neutron coincidence collar was built in the institute, for measuring total (gross) and coincidence count rates [3]. The first model was operating with 9 $^{3}$He tubes, 9 preamplifier-amplifier-discriminator chains, a JSR-11 shift register (loan from IAEA), and a notebook computer using standard Agency software (INCC code). Detection efficiencies of the two moderator configurations were 2.8 and 8.8 % for Pu-Be sources [4, 5]. Demonstration of the instrument and method, including measurements on selected sources, was presented to the IAEA, as a part of the Hungarian support programme to IAEA safeguards, in 2005.

Since then a new, upgraded detector system was designed and built, with 14 $^{3}$He tubes, new, faster electronic units, and a JSR-14 shift register (loan from IAEA). Different detection efficiencies (5, 9.6, and 11.3 %) can be selected using three moderator configurations, depending on the neutron output of the source to be assayed. These configurations were optimized with the aid of Monte Carlo simulation.

Since sources of certified Pu content were not available, calibration was carried out by using the results of measurements of a series of sources by a calorimeter, provided by the Institute for the Protection and Security of the Citizen (EC JRC IPSC, Ispra, Italy) in an informal collaboration. An additional, independent NDA method was also developed using pure gamma spectrometry, without neutron measurement [6, 7]. Furthermore, a method of using pure neutron measurements only was developed as well, without $\gamma$-spectrometry at all. Based on total (T) and coincidence (R) count rate measurements, this so-called R/T method relies on a correlation established between the ratio R/T and the Pu content [6, 8]. A demonstration of the upgraded equipment and the new methods to IAEA took also place in 2006.

In this report the results of the work is summarized. Source strengths range from $10^4$ to $10^7$ neutron/s, while the nominal Pu contents range from 0.1 to 178 g. The Pu–Be neutron sources are encapsulated in steel cylinders of 3–5.5 mm wall thickness. Their outer diameter and height vary from 10 to 35 and 19 to 45 mm, respectively. An additional encapsulation was necessary due to the expired garantee time of the original encapsulation as well as to the rusty surface of some of the capsules.

## 2. Measurement of the isotopic composition

A large area planar Ge detector (diameter 50 by 20 mm thick, resolution 688 eV FWHM at 122 keV) was applied. Spectra were taken for 10–50 min counting time at 50–200 cm source-to-detector distance, while the sources were taken out of their containers. The results show that isotopic composition varies in broad ranges. The $^{239}$Pu abundance amounts to from 75 up to 96%.

Pu isotope mass fractions and the ratio of Am to total Pu mass were evaluated by HRGS using the advanced commercial Multi-Group Analysis computer code MGA++ [9, 10], analyzing gamma- and X-rays below 300 keV. The evaluation is based on the peak-ratio technique.

## 3. Determination of the Pu content

### 3.1. Combined neutron-gamma method

Gross neutron output of the source, taken out of its container and inserted in the cavity of the collar, is measured for 2000–3000 s. The Pu content is calculated from neutron output and isotopic composition, relying on specific (alpha,n) yields adopted for individual Pu isotopes and Am, as:

$$m_{Pu} = \frac{N}{M \sum_i f_i g_i} \qquad (1)$$

where
$N$ is the neutron output,

$M$ is the multiplication in the source due to secondary (neutron-induced) reactions (see below),
$f_i$ is the abundance of the $i$–th (Pu and Am) isotope,
$g_i$ is the specific (alpha,n) yield of the $i$–th isotope.

Summation goes over all the isotopes, including that other than Pu as well. Since $f_i$ values are expressed in terms of percentage of the total Pu content, $\sum f_i$ exceeds 100% by the abundance of $^{241}$Am, and the formula gives the Pu content only (considered to be a "true" value in contrast with the nominal one).

Specific (α,n) reaction yield ($g_i$) values for the Pu (and Am) isotopes were determined by starting with the specific alpha activities from the literature, multiplied by n/alpha ratios, which convert activity to neutron output. These ratios are sensitive to alpha energies, and thus are different for individual isotopes. The products obtained in this way are maximum attainable values. They can be much less, if the Pu and Be constituents are incompletely dispersed and mixed in the source material upon production of the sources. Thus, the products of the two factors are to be normalized. This was carried out by using the results of calorimetric measurements. The heat output was measured for 19 PuBe sources by the ANTECH Small Sample Calorimeter Model 601 provided by JRC IPSC, Ispra [11]. The instrument was previously calibrated in the PERLA laboratory of the IPSC using certified reference materials. By combining heat results with isotopics determined by gamma spectrometry, Pu masses were determined, relying on specific heat values from the literature. Using such a calibration, normalized specific (alpha,n) yields ($g_i$) were obtained, as follows (yields of $^{241,242}$Pu are neglected):

$^{238}$Pu: $2.89\times10^{7}$ n/g·s

$^{239}$Pu: $8.86\times10^{4}$ n/g·s

$^{240}$Pu: $3.25\times10^{5}$ n/g·s

$^{241}$Am: $5.76\times10^{6}$ n/g·s.

These figures are valid for sources produced in the late Soviet Union till September 1978. According to an informal notification from the manufacturer, sources produced later are of specific neutron yield twice as high. We had no opportunity to check this statement as yet.

The $^{241}$Pu and $^{242}$Pu specific yields, being of the order of $10^3$, negligible at the usual isotopic ratios, were not considered. Similarly, the neutron yield from spontaneous fission of isotopes of even mass number was neglected as well.

In addition to gross neutron (totals, singles) counts, coincidence (doubles) counts were also recorded. Real coincidences (reals, after subtracting accidental coincidences) are due to secondary reactions, i.e. neutron-induced fission of the Pu isotopes (self-multiplication) and the $^{9}$Be(n,2n)$^{8}$Be reaction. Contribution of spontaneous fission neutrons, as treated in Refs. [3-5], can be estimated to be negligible.

The multiplication in the source itself was taken into account by the correction factor $M$. This correction may amount to 15–20 % for the strongest sources, and can be determined by using coincidence measurements or Monte Carlo calculations. The results were practically the same by the two methods. An equivalent way is the use of the analytical formula

$$M = 1+0.034\log m_{Pu} +0.0153\log^2 m_{Pu} +0.00283\log^3 m_{Pu} \quad (2)$$

and, if necessary, of iteration (usually 2 steps are sufficient).

Uncertainty of the Pu mass determined by calorimetry was mainly due to the error in determining the isotopics, which was in general taken to be 3 - 4 %, while the systematic error of neutron output was given originally as 10 %. Nevertheless, the standard deviation among the measured sources of the same nominal (declared) neutron output was 2-3 % only, therefore the precision of the combined method may well approach 5–6 %, even though the absolute value may differ from the real Pu mass by 8–10 %.

### 3.2. Pure gamma spectrometry

Pu masses were also determined by pure HRGS, without neutron measurements. The method relies on absolute intensity measurements of $^{239}$Pu photopeaks, applying attenuation correction [6, 7], and taking into account the $^{239}$Pu abundance determined by $\gamma$-spectrometry.

The source is taken out of its container and the 375 and 413 keV photopeak areas of $^{239}$Pu are measured by a large planar Ge detector for 10 – 20 min counting time in a far-field geometry (at 50 – 200 cm distance from the source, depending on its size and strength). Attenuation correction is applied, assuming a parallel beam falling on the detector surface and that the Pu-Be source has a cylindrical shape.

The abundance of $^{239}$Pu $f_{239}$ is determined from the same gamma spectrum. The precision of the method is about 5%. An accuracy of 6-10 % can be attained, depending on source strength, measurement time, knowledge of the source diameter-to-length ratio, and wall thickness of the source.

### 3.3 Pure neutron counting ("*R/T* method")

In addition to total neutron counting, coincidence counting can be exploited as well. Pure neutron measurements are carried out in this way, without $\gamma$-spectrometry at all. Based on total (*T*) and coincidence (*R*) count rates, this so-called *R/T* method [6, 8] relies on a correlation established between the ratio *R/T* and the Pu content determined by calorimetry, using the graphs or the corresponding formulae below (Fig. 1).

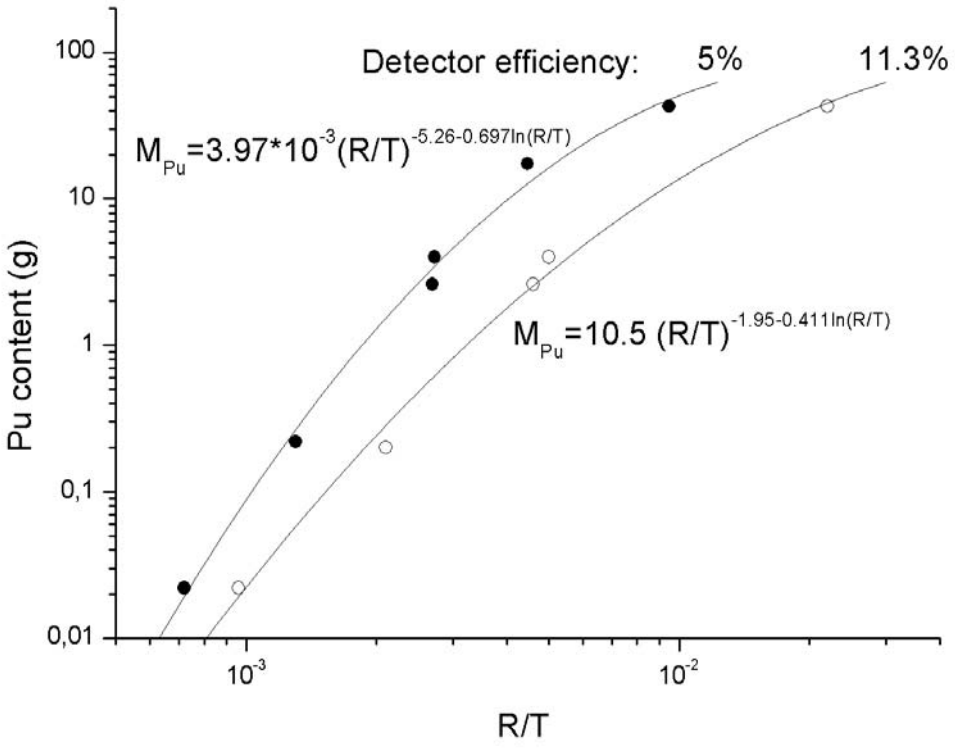


**Fig. 1.** Calibration of the *R/T* method against calorimetry

The couple of parameters $[R_{norm}, T_{norm}=N]$ normalized to 100 % efficiency do not depend on the parameters of the neutron coincidence collar. An empirical formula, fitted to calorimetric results

$$m_{Pu} = 0.225\left(R/T\right)_{norm}^{2.65-0.23\ln(R/T)norm} \quad (3)$$

gives Pu content by an accuracy of within 23 % on average. This method may need longer counting time than the two previous ones.

Comparison of results obtained by various methods is seen for 7 representative sources in Table 1.

| Declared neutron output (n/s) | $^{239}$Pu fraction (%) | Pu content (g) | | | | |
|---|---|---|---|---|---|---|
| | | Nominal | Combined n-gamma | Pure gamma spectrometry | Calorimetry | Pure neutron counting |
| $1.1\times10^4$ | 83.15(0.83) | 0.18 | 0.022(0.003) | 0.021(0.003) | 0.0223(0.0041) | 0.023(0.003) |
| $1.1\times10^5$ | 79.76(0.80) | 1.8 | 0.22(0.01) | 0.225(0.004) | 0.224(0.007) | 0.23(0.03) |
| $2.68\times10^5$ | 95.21(0.19) | 4.0 | 2.48(0.13) | 2.55(0.10) | 2.62(0.08) | 2.6(0.3) |
| $2.26\times10^6$ | 76.92(0.82) | 37 | 3.93(0.23) | 4.0(0.3) | 4.0(0.12) | 4.4(0.7) |
| $5.58\times10^6$ | 76.17(1.0) | 45 | 9.62(0.63) | 9.5(0.5) | 10.4(0.31) | 7.33(2.5) |
| $1.1\times10^7$ | 75.7(0.91) | 178 | 17.05(1.15) | 17.8(1.6) | 17.3(0.52) | 15.3(2.9) |
| $5.27\times10^6$ | 94.91(0.19) | 85 | 42.1(3.0) | 44.0(1.5) | 44.3(1.3) | 46.9(4.7) |

**Table 1:** Main characteristics of 7 representative Pu-Be sources. Pu contents are determined by various methods (errors are in brackets)

Particularly, the couple $[R_{norm}, T_{norm}=N]$ can be used for estimating both the isotopic composition and the total Pu content ("R/T-T method"):

$$(R/T)_{norm} = 1.787\left(\frac{N/M}{a+bf_{239}}\right)^{0.44+0.024\ln\left(\frac{N/M}{a+bf_{239}}\right)} \quad (4)$$

$$m_{Pu} = \frac{N/M}{a+bf_{239}}, \quad (5)$$

where the values *a* and *b* are constant and found to be (21.4±2.1)x$10^5$ and -(0.214±0.01)x$10^5$ n/s·g, respectively [12].

Errors of the "*R/T-T* method" are about 2-3% for $M_{239}$ and 15-20% for the Pu content.

Results obtained by using Eq. (4) are plotted in Fig.2 for various $f_{239}$-s, which were estimated from the function $[(R/T)_{norm}, T_{norm}=N]$ by graphical method.

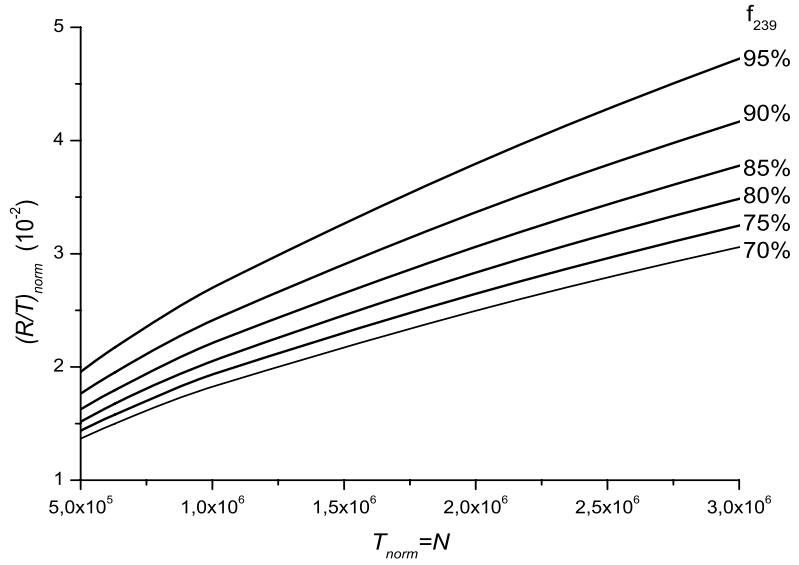


**Fig. 2.** $(R/T)_{norm}$ for various $^{239}$Pu abundances as a function of $T_{norm}=N$

## 4. Serial measurements, re-encapsulation, and disposal

In a comprehensive measurement programme, neutron output, isotopics, and Pu content of 76 Pu-Be sources, stored in the institute, were determined. As a result, it has been turned out that facility and State inventories are based on incorrect, highly overestimated values. Nominal Pu masses have been overestimated even by an order of magnitude in some cases (where the $^{239}$Pu abundance is about 75 % of the total Pu content). The measurements resulted in a total amount 563±15 g Pu, in contrast with 2050 g according to the sum of declared nominal values in the files. A new inventory of the Pu content of all the Pu-Be sources in the country is planned to be taken as well. The method is offered for routine IAEA use and also to other countries facing similar problems.

The measured sources were additionally encapsulated. In the course of this, the used Pu-Be sources in their present form – i.e. in their old capsules - were sealed into stainless steel (KO-36 type) holders with a wall thickness of 1.5–2.0 mm. The holders were sealed by using Argon gas-protected welding. The new holders (diam. 22.2–42, length 50–147 mm) contain 1–4 old Pu-Be sources in order to decrease the required storage room for the sources in the final disposal. After the re-encapsulation, the batch numbers of the old sources (as used in safeguards accountancy) were engraved onto the surface of the new holders to help their future identification.

The newly encapsulated sources were placed in special containers for shipping and storing according to the ISO 9001 quality assurance system. The containers were shipped to the radioactive and nuclear waste disposal facility at Püspökszilágy, Hungary, where they remained accessible to identification and verification for authorities.

A new inventory of the Pu content of all the Pu-Be sources in the country is planned to be taken as well. The method is offered for routine IAEA use and also to other countries facing similar problems.

## 5. Acknowledgements

This work was supported by the Hungarian Atomic Energy Authority under contracts No. OAH-ÁNI-ABA-16/01, -04/02, -04/03 and -01/04. IAEA is gratefully acknowledged for providing loan of the shift registers. IPSC Ispra is also thanked for making available the calorimeter and the neutron coincidence collar Type JCC-13 for test purposes.

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# "Multi-Dynamic" Isotope Ratio Measurements for Uranium and Plutonium using the "Multiple Ion Counting" Detection System of the TRITON TIMS at IRMM

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***Keywords:*** Uranium isotope measurements, Multiple Ion Counting, Multi-Dynamic Measurements

## *Abstract:*

*A "Multiple Ion Counting" (MIC) detection system installed into a thermal ionization mass spectrometer (TIMS) allows the simultaneous detection of up to seven small ion beams with currents of $10^{-19}$ – $10^{-14}$ Ampere in ion counting mode, corresponding to count rates of 1–60.000 cps (counts per second). In order to circumvent complicated inter-calibration routines for the given set of 7 ion counters the principle of multi-dynamic measurements was implemented in combination with the "Multiple Ion Counting" (MIC) system.*

*The multi-dynamic measurement procedure was applied to diluted samples of isotope reference materials for uranium and plutonium with concentrations at the ppb and the ppt level, respectively. The results for uranium measurements on samples IRMM-186 and NBL-U500 clearly show an improved precision by a factor of ca. 3 for the multi-dynamic measurement procedure compared to measurements carried out using one single ion counter in peak-jumping mode. When using the "Multiple Ion Counting" (MIC) system, the multi-dynamic procedure is also superior compared to the static total evaporation technique, as demonstrated by plutonium measurements on samples of NBL-CRM137.*

*As a conclusion, the multi-dynamic procedures provide improved precision and accuracy compared to previous TIMS measurement techniques. The multi-dynamic measurement technique in combination with the "Multiple Ion Counting" (MIC) system of the TRITON TIMS is suitable for the isotopic analysis on low-level uranium and plutonium samples as well as single uranium or plutonium oxide particles for nuclear safeguards purposes, and will be applied also for the verification analysis of reference particles made of $UF_6$ as produced by IRMM.*

## *Introduction*

A new detector system designed for isotope ratio mass spectrometers provides improved precision on measurements of actinide samples with very low amounts (< $10^{-10}$ grams) of analyte. An array of continuous dynode electron multipliers (CDEMs) has been installed on a Thermo Electron Triton thermal ionization mass spectrometer acquired by IRMM in 2004. These ion counters are miniaturized continuous dynode electron multipliers. They can be readily installed to replace individual Faraday cups in a multi-detector mass spectrometer or bundled together and located along the detector plane with a set of Faraday cups. On the IRMM Triton mass spectrometer, nine Faraday cups, one conventional discrete dynode electron multiplier, and seven CDEMs were installed.

The size of the CDEMs is identical to the standard Faraday cups, and thus the ion counters can be aligned with unit mass spacing for the measurement of high mass elements such as U and Pu. Six of the small ion counters were bundled together and positioned on the low mass side of the L4 Faraday cup, see Fig.1. One additional ion counter was positioned on the high mass side of the L4 Faraday cup. This arrangement (the so called “Goldberg-package”, see [1]) allows for simultaneous measurement of either all uranium (including $^{233}U$) or plutonium (including $^{244}Pu$) isotopes, also a combination of the two. Compensation for small mass dispersion differences at unit mass spacing of U, Pu, is readily achieved by the use of a mass dispersion quadrupole zoom lens.

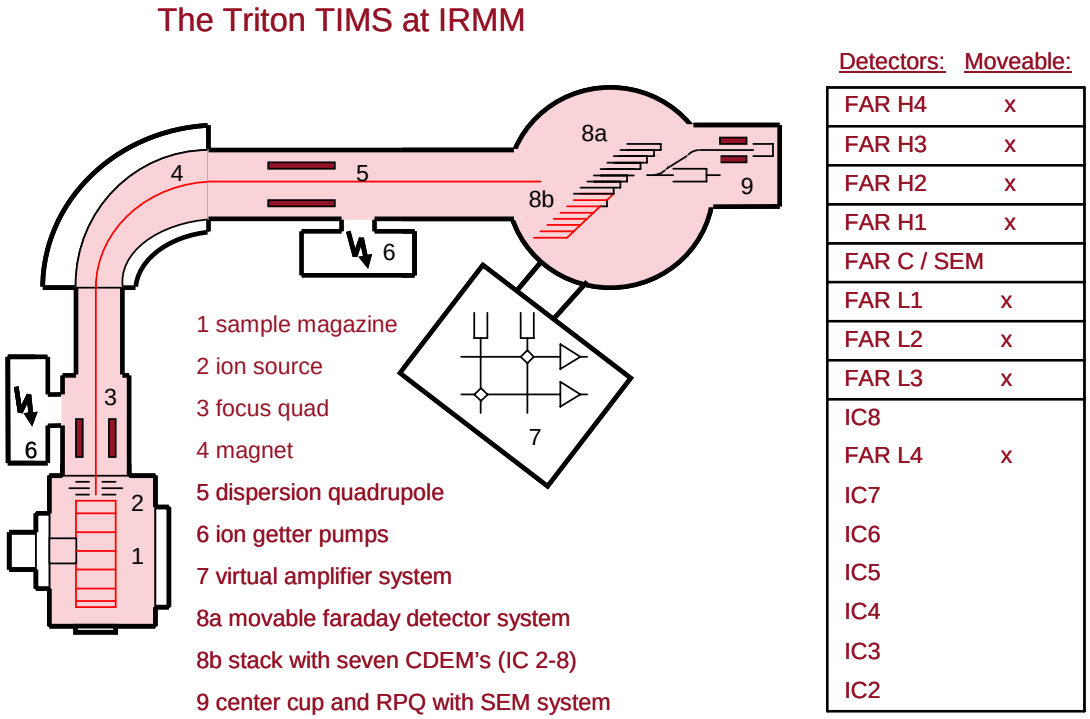


Fig.1: The configuration of the TRITON TIMS at IRMM

The advantage of multiple ion counting is the simultaneous collection of several isotopes of a given element [1, 2, 3]. It overcomes many of the problems such as transient signal variation in sample emission and ionization, which would significantly reduce the attainable precision of a single collector measurement. For a given sample, a multiple ion counting measurement makes use of a greater number of ions counted for each isotope compared to a peak-jumping measurement using only a single ion counting detector and therefore provides improved counting statistics by a factor of two or more. In addition, multiple ion counting eliminates the need for drift correction algorithms that are required in single collector measurements and thus eliminate a potential source of systematic error.

Acceptance tests performed both at the factory and after the installation at IRMM indicate that the CDEM ion counters exhibit high counting efficiency and a dark noise of less than 10 counts per minute. The multiple ion counters have sufficient stability to yield

a repeatability of 0.5% RSD on the $^{234}U/^{238}U$ and $^{236}U/^{238}U$ ratios in uranium samples of CRM U500. This precision was obtained using the multi-dynamic measurement procedure. The "Multiple Ion Counting" system at IRMM, its installation and first test measurements have been described in detail in [3]. This paper is focused on the demonstration of the multi-dynamic procedure for low-level U and Pu isotope measurements.

## *The "Multi-dynamic" Mass Cycles for Uranium and Plutonium*

The main characteristics of the multi-dynamic measurement technique is the fact that the calibration factors of all ion counters are mathematically eliminated from the mathematical algorithms for the so-called minor ratios, e.g. for uranium this applies to the $^{234}U/^{235}U$ and $^{236}U/^{235}U$ ratios, the algorithms are given in [3]. But it has to be emphasized that the calibration factors cannot be eliminated for the measurement of the so-called major $^{235}U/^{238}U$ (or $^{238}U/^{235}U$) ratios. Therefore the $^{235}U/^{238}U$ is measured in a peak-jumping type of measurement, using the $^{235}U$ measured in step 1 and $^{238}U$ measured in step 4, both detected by ion counter IC6 as shown in Tab. 1 below. The k-factor for mass fractionation correction of the $^{235}U/^{238}U$ ratio has to be determined externally using the known isotope ratios of the IRMM-186 isotopic standard, to be measured using the same technique on the same sample turret. The mass fractionation correction for the minor ratios is performed internally using the major ratio, as explained in [3]. As also explained in [3], for the minor ratios of a given sample a background correction has to be done based on the measured minor ratios of the standard.

In case of plutonium, the minor ratios are $^{241}Pu/^{239}Pu$ and $^{242}Pu/^{239}Pu$. The $^{238}Pu/^{239}Pu$ minor ratio is not taken into account because in most cases of low-level Pu samples a strong interference from $^{238}U$ hampers a reasonable measurement of the $^{238}Pu$ abundances. The $^{240}Pu/^{239}Pu$ is the major ratio; it is measured in peak-jumping mode using IC3 in steps 1/2, using IC4 in steps 2/3 and using IC5 in steps 3/4, see Tab. 2.

| Channel: | IC 2 | IC 3 | IC 4 | IC 5 | IC 6 | IC 7 |
|---|---|---|---|---|---|---|
| Step: | | | | | | |
| 1 | | | | 234 | 235 | 236 |
| 2 | | | 234 | 235 | 236 | |
| 3 | | 234 | 235 | 236 | | 238 |
| 4 | 234 | 235 | 236 | | 238 | |

Tab.1: The multi-dynamic mass cycle for uranium

| Channel: | IC 2 | IC 3 | IC 4 | IC 5 | IC 6 | IC 7 |
|---|---|---|---|---|---|---|
| Step: | | | | | | |
| 1 | 239 | 240 | 241 | 242 | | 244 |
| 2 | | 239 | 240 | 241 | 242 | |
| 3 | | | 239 | 240 | 241 | 242 |
| 4 | | | | 239 | 240 | 241 |

Tab.2: The multi-dynamic mass cycle for plutonium

### *Results*

Uranium isotopic measurements have been carried out for diluted samples of the isotope reference materials IRMM-186 and NBL-U500, the filament load was ca. 1ng. Replicate measurements were performed in 2 ways, first using the multiple ion counting system in combination with the multi-dynamic measurement technique and second using one single ion counter in peak-jumping mode. The results are shown in Tab. 3 and Tab. 4. They clearly show an improvement in precision for the multi-dynamic technique by a factor of about 3. This is simply due to counting statistics.

| IRMM-186 | Multiple Ion Counting + Multi-Dynamic $^{234}U/^{238}U$ | Single ion counter - Peak-Jumping $^{236}U/^{238}U$ |
|---|---|---|
| AVERAGE | 0.0002969 | 0.0003009 |
| SD | 0.0000023 | 0.0000064 |
| RSD | 0.76% | 2.1% |

Tab.3: The multi-dynamic measurements for IRMM-186 in comparison with peak-jumping measurements using one ion counter. The improvement in precision is a factor of ca. 3.

| NBL-U500 | Multiple Ion Counting + Multi-Dynamic $^{236}U/^{238}U$ | Single ion counter - Peak-Jumping $^{236}U/^{238}U$ |
|---|---|---|
| AVERAGE | 0.0015255 | 0.001519 |
| SD | 0.0000065 | 0.000018 |
| RSD | 0.43% | 1.2% |

Tab.4: The multi-dynamic measurements for U500 in comparison with peak-jumping measurements using one ion counter. The improvement in precision is a factor of ca. 3.

Isotopic measurements for plutonium have been carried out for diluted samples of the isotope reference material NBL-CRM-137, the filament load was ca. 5pg. Replicate measurements were performed in 2 ways both using the multiple ion counting system: first by applying the multi-dynamic measurement technique and second in static total evaporation mode. The results are shown in Tab. 5 and Tab. 6. They clearly show an improvement in precision for the multi-dynamic technique by a factor of about 4-5. This

is due to the *internal* calibration of the ion counting system achieved within the multi-dynamic measurement technique, as opposed to the *external* calibration which is done on a filament by filament basis for the static total evaporation measurements. Apparently the counting efficiencies for the various ion counters of the MIC system change significantly from one sample filament to another, the counting efficiencies seem to depend on the different ion beam focussing conditions for each filament. This type of variability is circumvented by using the multi-dynamic technique.

| NBL-CRM-137 | Multiple Ion Counting + Multi-Dynamic $^{241}Pu/^{239}Pu$ | Multiple Ion Counting + Total Evap. $^{241}Pu/^{239}Pu$ |
|---|---|---|
| AVERAGE | 0.011050 | 0.01088 |
| SD | 0.000038 | 0.00023 |
| RSD | 0.34% | 2.1% |

Tab.5: The multi-dynamic measurements for NBL-CRM-137 in comparison with static total evaporation measurements. The improvement in precision is a factor of ca. 4-5.

| NBL-CRM-137 | Multiple Ion Counting + Multi-Dynamic $^{242}Pu/^{239}Pu$ | Multiple Ion Counting + Total Evap. $^{242}Pu/^{239}Pu$ |
|---|---|---|
| AVERAGE | 0.015601 | 0.01529 |
| SD | 0.000048 | 0.00022 |
| RSD | 0.31% | 1.4% |

Tab.6: The multi-dynamic measurements for NBL-CRM-137 in comparison with static total evaporation measurements. The improvement in precision is a factor of ca. 4-5.

## *Conclusions*

The multi-dynamic measurement technique was combined with the multiple ion counting system of the Triton TIMS for measurements of ng-size uranium samples and pg-size plutonium samples. The multi-dynamic technique provides the following features:

- The inter-calibration of multiple ion counting system is achieved *internally* using the ion beams of the sample.
- The multi-dynamic technique provides a significant improvement in precision compared to
    a. peak-jumping measurements (due to counting statistics), as shown by an example of uranium measurements.
    b. static total evaporation measurements (due to internal calibration), as shown by an example of plutonium measurements.

As a conclusion, the multi-dynamic technique in combination with the multiple ion counting detection system is the preferred measurement technique for isotopic measurements of U and Pu in low-level samples. Using the multiple ion counting system in static total evaporation mode does not provide the full benefit of this detection system. To take full advantage of the multiple ion counting system, an *internal* calibration of the ion counters is preferred, which is only provided by the multi-dynamic technique.

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# Results for the "REIMEP 18" Inter-laboratory Comparison Campaign for the Measurement of Uranium Isotope Ratios

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***Keywords:*** REIMEP, Uranium isotopes measurements

## *Abstract:*

*Inter-laboratory comparisons are organized for a variety of sample types and elements in order assess the performance of isotopic measurements on a worldwide level.*

*The REIMEP 18 ("Regular International Measurement Evaluation Program) campaign for the measurement of uranium isotope ratios in nitric acid was started in December 2005. Four samples of 2.5mg uranium, ranging from depleted up to slightly enriched uranium, are sent to more than 85 participating laboratories, originating mainly from the nuclear safeguards and the isotope geochemistry area, and using a variety of techniques such as alpha-spectrometry, TIMS, ICP-MS, AMS, RIMS, etc. As observed during several REIMEP campaigns organized during the past 10 years, REIMEP 18 is designed to show the present state of uranium isotope measurements and gives the opportunity for participating laboratories to evaluate their own performance, to identify possible problems and to improve their own measurement procedures. Due to instrumental improvements in measurement techniques and instrumentation continuously going on over the years, measurement campaigns such as REIMEP are an important and very much appreciated way to achieve an ongoing careful quality control on an international level.*
*Certification measurements at IRMM were performed using recently upgraded techniques for high precision and high accuracy uranium isotope ratio measurements. $^{235}U/^{238}U$ measurements were performed using a $UF_6$ gas source mass spectrometer, calibrated using synthetic isotope mixtures. $^{234}U/^{238}U$ measurements, even down to values of $5.5x10^{-5}$ (natural equilibrium value) were performed on a TRITON TIMS, using $10^{12}$ Ω Faraday cup amplifiers for the detection of $^{234}U$ in order to improve the signal-to-noise-ratio, so without the need to use ion counting. $^{236}U/^{238}U$ measurements were performed on a TRITON TIMS using a procedure in which $^{236}U$ was detected using an ion counter and which has been validated using IRMM's special synthetic mixtures with $^{236}U/^{238}U$ ratios of $10^{-6}$, $10^{-7}$ and $10^{-8}$.*

*Details of the sample preparation and certification will be given as well as comparative results of the measurements made by all ca. 70 participating laboratories worldwide. The results of the REIMEP 18 campaign confirm in general the excellent capability of laboratories in measuring isotopic abundances of uranium, but also identify some problems regarding the measurements of the minor isotope ratios $^{234}U/^{238}U$ and $^{236}U/^{238}U$.*

## *Introduction*

Measurements to determine the isotopic composition of uranium samples are done in a variety of industrial and scientific areas. Firstly, because of the industrial use of uranium

as fuel in nuclear power reactors, the relative isotopic enrichment of the fissile uranium isotope $^{235}U$ has to be quantified within each section of the nuclear fuel cycle, e.g. the enrichment facilities, the power plant operating facility, the reprocessing plant and finally the waste handling facility. The isotopic composition of uranium material going through the nuclear fuel cycle is subject to careful verification analyses by national as well as international nuclear safeguards authorities such as the IAEA and EURATOM inspectorates. Secondly, the uranium isotopic composition is also measured in many scientific disciplines, such as geochemistry for disequilibrium studies or geochronology.

Due to the scientific and political relevance of nuclear isotopic measurements all safeguards laboratories need a reliable quality management system to ensure their measured values are acceptable. Nuclear analytical laboratories are required to demonstrate their measurement capability on a regular timely basis. One way of demonstrating measurement capability is to participate in inter-laboratory comparisons. For this reason, IRMM has organized quality control campaigns for measurements of uranium and plutonium for safeguards and fissile material control for more than 20 years. The REIMEP programme (Regular European Inter-laboratory Measurement Evaluation Programme) serves as a tool for laboratories to demonstrate their abilities to measure uranium and plutonium isotopic ratios in a variety of sample forms, chosen where possible to be typical of fissile material samples commonly found in the nuclear industry and controlled by nuclear safeguards authorities. Previous REIMEP campaigns have included samples such as uranium oxide, uranium in nitric acid, uranium in the form of $UF_6$ [1, 2], plutonium oxide, and others.

For the REIMEP 18 campaign the uranium samples were provided in nitric acid solution, a matrix that is easy to handle and does not require extensive sample preparation steps, such as separation from other elements or purification. For this reason the REIMEP 18 campaign focussed on the 'pure' instrumental part of uranium isotope measurements by the technique the laboratory applied.

The mass-spectrometric certification measurements for uranium isotope ratio measurements for the REIMEP 18 campaign were performed at IRMM applying the latest knowledge of measurement methodology. This led to state of the art precision and accuracy in isotopic measurements. Four samples of depleted to low-enriched uranium were selected from the IRMM stock. The original uranium samples were in $UF_6$ form. They were certified for the major ratio $n(^{235}U)/n(^{238}U)$ using a Varian MAT511 $UF_6$-gas source mass spectrometer (GSMS), calibrated using certified materials traceable to synthetic isotope mixtures. The samples in $UF_6$ form were hydrolyzed and calcined in order to obtain the uranium in oxide form ($U_3O_8$). The oxides were dissolved in nitric acid to obtain the batch solutions for REIMEP 18 A - D. In order to verify the certified major ratios $n(^{235}U)/n(^{238}U)$, thermal-ionization mass-spectrometer (TIMS) measurements were performed using the “Modified Total Evaporation” (MTE) technique as described in [3, 4].

The minor uranium isotope ratios, $n(^{234}U)/n(^{238}U)$ and $n(^{236}U)/n(^{238}U)$ were then measured and certified using a Triton TIMS. The method is described in detail in [10, 11]. All $n(^{234}U)/n(^{238}U)$ and $n(^{236}U)/n(^{238}U)$ ratios higher than ca. $5x10^{-5}$ were measured using Faraday collectors only, with current amplifiers that were equipped with $10^{12}\Omega$ resistors to improve the signal to noise ratio. All $n(^{236}U)/n(^{238}U)$ ratios below $5x10^{-5}$, which only applied to samples REIMEP 18 A and D, were measured using an SEM (secondary electron multiplier) in combination with an energy filter for improved abundance sensitivity. This was inter-calibrated against the Faraday cups using the $^{234}U$ beam. The

batch solutions for REIMEP 18 A - D were dispensed into 100 ampoules each in order to have a sufficient supply for all interested participants. Each of the ampouled samples contained 2.5 mg uranium in 0.5 mL 0.5 M nitric acid solution. The sample amounts were chosen in order to achieve a total alpha activity of less than 1000 Bq for each set of four samples of REIMEP 18 A - D, which allowed the sample sets to be shipped as non-nuclear material. This was a significant advantage for a large number of participating laboratories, because extensive administrative work related to nuclear transport requirements could be avoided.

## *Results and Discussion*

Results for the $n(^{234}U)/n(^{238}U)$, $n(^{235}U)/n(^{238}U)$ and $n(^{236}U)/n(^{238}U)$ ratios for samples REIMEP 18 A - D are presented in Figures 1-4. In each graph the various measurement techniques used are indicated. A general observation for all ratios and all the samples is that the spread among the data increases with decreasing ratio values; this is simply related to the ion beam intensities, counting statistics and/or amplifier noise. For all techniques the spread of the data is significantly (at least 2-3 times) higher than the certified range, which is displayed in grey as $\pm ku_C$ with coverage factor k = 2.

The certified $n(^{235}U)/n(^{238}U)$ ratios of samples REIMEP 18 A - D lie between ca. 0.004 (depleted uranium) and 0.035 (low enriched uranium, e.g. see Fig.1.) and cover a range typical for nuclear safeguards samples.

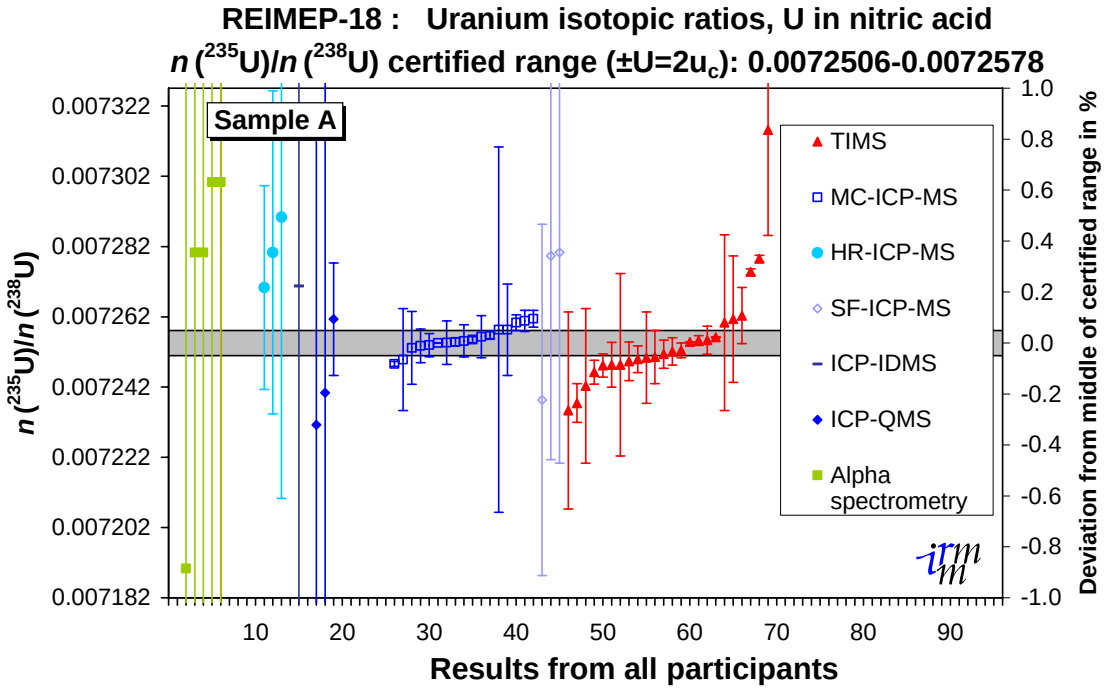


Figure 1: Results for the $n(^{235}U)/n(^{238}U)$ ratio for REIMEP 18 A

From the $n(^{235}U)/n(^{238}U)$ results from all participants the following observations can be made:

1. The data spread depends on the applied techniques: multi-collector inductively–coupled-plasma mass spectrometry (MC-ICP-MS) and TIMS show the smallest spread. For alpha spectrometry the results are between ±1% of the certified value only for sample REIMEP 18-A which has an enrichment close to natural. For all other samples the deviations are more than 1% and the data are therefore not shown on the graphs.
2. The uncertainties reported for many of the MC-ICP-MS and TIMS results seem to be quite small and appear be underestimated. One possible reason might be the fact that several participants used natural uranium samples as a standard for the mass fractionation correction by using the consensus value of $n(^{238}U)/n(^{235}U)=137.88$ as "reference value". Although this number is well known and established in the literature, it is not a certified value and moreover, it does not have any (certified) uncertainty associated with it. As a consequence, the uncertainty contribution of this standard sample used for mass fractionation correction, which is in many cases the dominant contribution, is missing within the uncertainty calculation for the corrected ratio of the sample. This neglect can lead to a dramatic underestimation of the uncertainties; in several cases this neglect might have even caused an apparent deviation of a measured ratio from the certified value.

The certified $n(^{234}U)/n(^{238}U)$ ratios of samples REIMEP 18 A - D lie between ca. 0.000055 (close to natural uranium, see Fig.2.) and 0.00035 (slightly enriched uranium) and cover a range typical for nuclear safeguards samples.

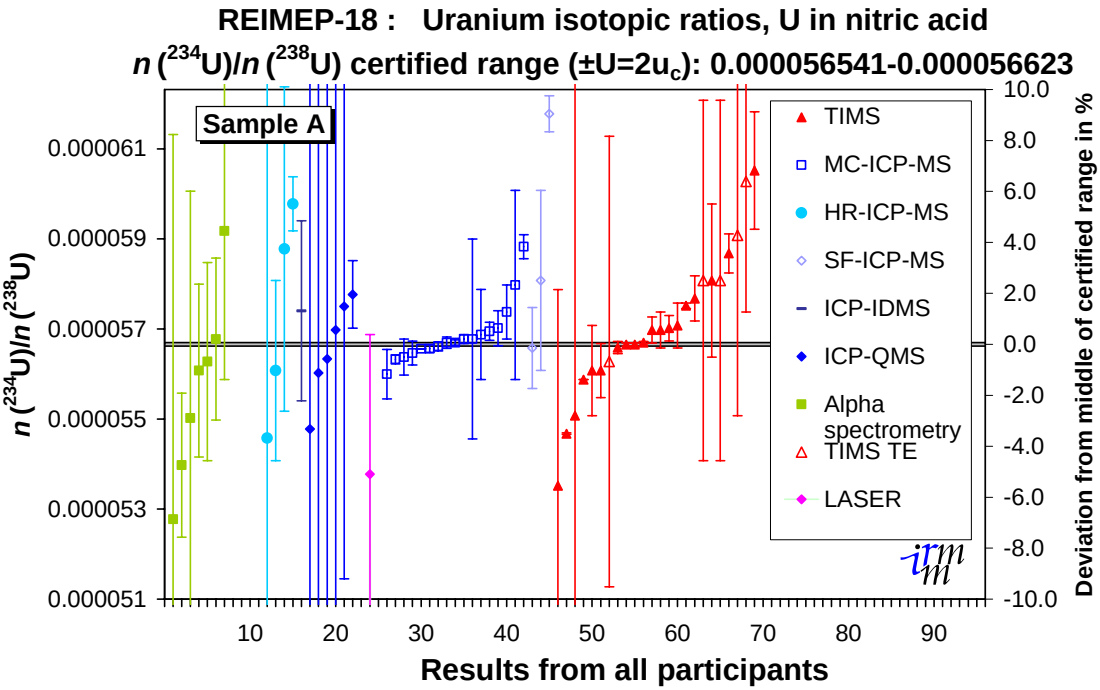


Figure 2: Results for the $n(^{234}U)/n(^{238}U)$ ratio for REIMEP 18 A.

From the $n(^{234}U)/n(^{238}U)$ results the following observations can be made:

1. The data spread depends on the applied technique. Measurements done using MC-ICP-MS show the smallest spread, followed by TIMS, then other ICP-techniques, isotope selective laser ionization mass spectrometry and alpha spectrometry.
2. For $n(^{234}U)/n(^{238}U)$ ratios smaller than about 0.0001, which applies to the samples REIMEP 18-A and REIMEP 18-D, the TIMS-TE results (TE = total evaporation) seem to be on average higher than the expected value. This is probably due to a neglect of the peak tailing correction to be done for the tailing of the major ion beams of $^{235}U$ and $^{238}U$. But this deviation is not as significant as found and described below for results for the $n(^{236}U)/n(^{238}U)$ ratios.

The certified values of $n(^{236}U)/n(^{238}U)$ for the REIMEP 18 samples lie between ca. $10^{-8}$ (closest to natural uranium, see Fig. 3) and 0.001 (similar to processed uranium, see Fig. 4).

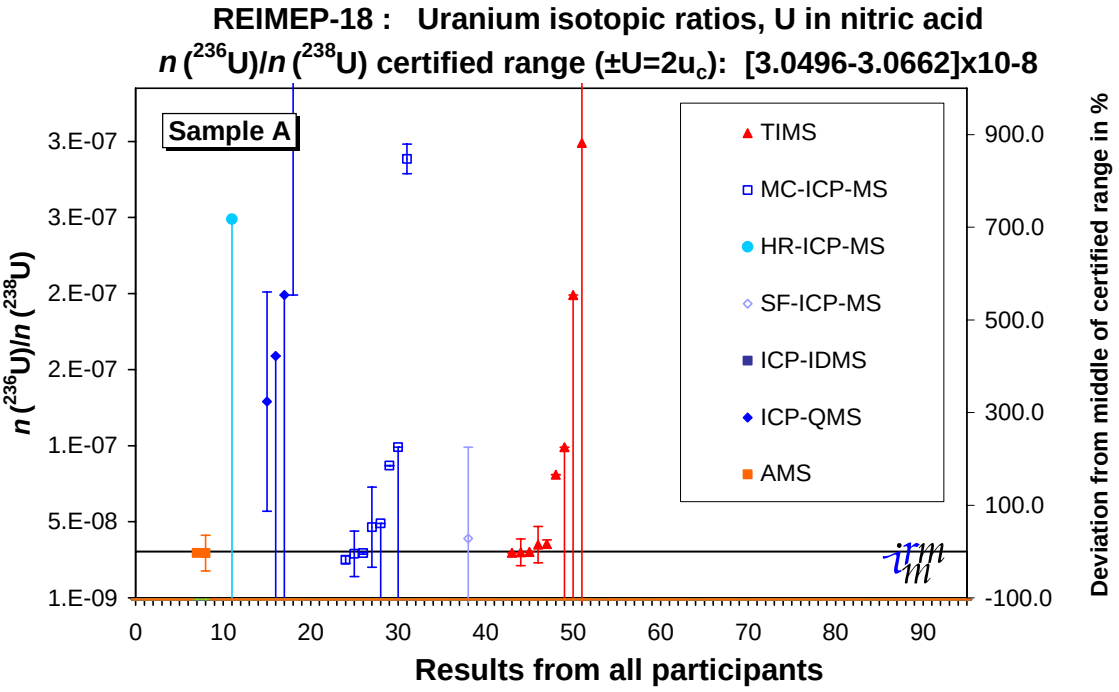


Figure 3: Results for the $n(^{236}U)/n(^{238}U)$ ratio for REIMEP 18 A.

From the $n(^{236}U)/n(^{238}U)$ results the following observations can be made:

1. The data spread depends on the techniques applied and on the order of magnitude of the ratio. For samples REIMEP 18 B and C with $n(^{236}U)/n(^{238}U)$ >0.0001, measurements performed using MC-ICP-MS show the smallest spread, followed by TIMS, other ICP-techniques and alpha-spectrometry. For samples REIMEP 18 D and A, with ratios of the order of $10^{-7}$ and $3x10^{-8}$, AMS (only 2 results) and TIMS show the smallest spread, followed by the ICP-MS techniques.

2. For samples REIMEP 18 B and C with ratios of 0.0003 and 0.001 respectively, most of the TIMS-TE results (TE = total evaporation) are significantly higher than the expected value. These deviations are very probably due to neglecting the peak tailing correction due to the large ion beam at mass 238. More care has to be taken for accurate measurements of $n(^{236}U)/n(^{238}U)$ when measured in simple static total evaporation mode.

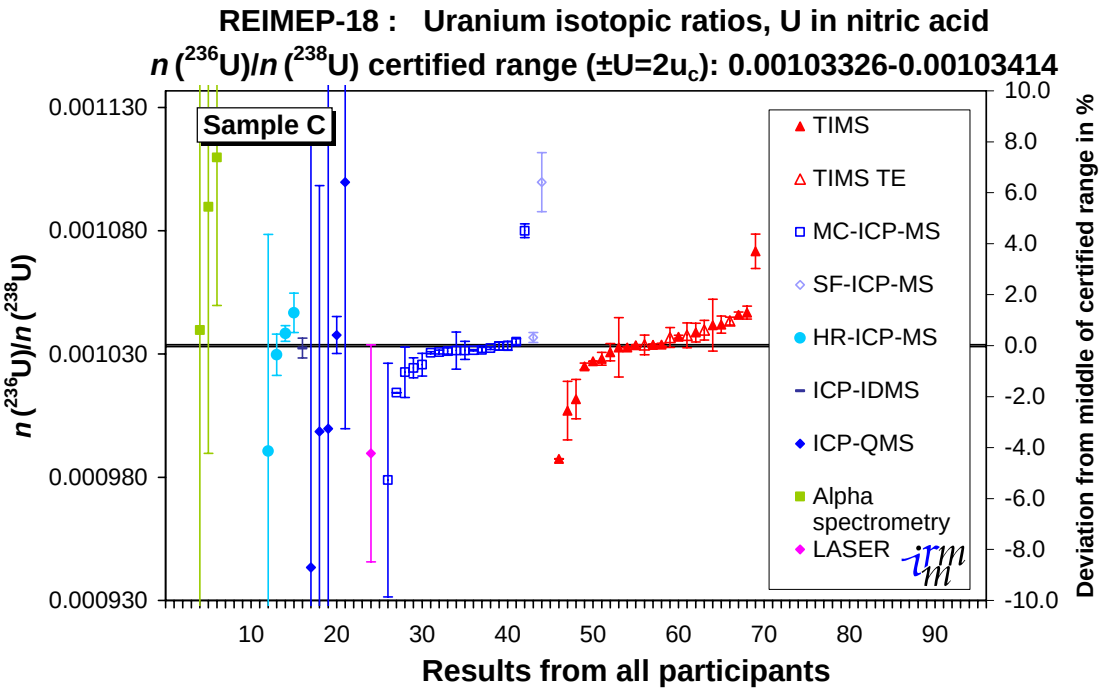


Figure 5: Results for the $n(^{236}U)/n(^{238}U)$ ratio for REIMEP 18 D

## *Conclusions*

The REIMEP 18 inter-laboratory comparison campaign for uranium isotope measurements was a success. The overall response of 85 registered and 71 actually participating laboratories was exceptionally high. The results of this campaign may therefore be considered as a representative picture of present uranium isotopic measurement capabilities for a broad range of disciplines and on a worldwide scale. The REIMEP 18 campaign samples are already being applied as quality control samples by several laboratories and even used to demonstrate the analytical performance in publications [5, 6]. From the results the following main conclusions can be drawn:

1. For measurements of uranium isotopic ratios a variety of different measurement techniques is in use. Most prominent is mass spectrometry, and among different types of mass spectrometry TIMS and various types of ICP-MS, especially MC-ICP-MS, are the most frequently used. Each technique has its limitations, e.g. alpha-spectrometry and isotope selective laser ionization mass spectrometry are preferentially used for certain isotope ratios or specific ranges of ratios. ICP-MS has

limited capabilities for measurements of ratios covering a large dynamic range, e.g. for $n(^{236}U)/n(^{238}U)$ ratios of the order of $10^{-7}$ and below. For this type of measurement TIMS and AMS show the best performance.

2. For TIMS measurements, the performance for routine nuclear safeguards measurements of the minor isotope ratios, e.g. using the total evaporation technique, does not reach the standard expected and has to be improved. It seems that the capabilities provided by modern TIMS instruments are not always fully applied in order to reach the best possible performance of TIMS. Plans for improvement have been proposed by IRMM, e.g. by an expanded implementation of the "modified total evaporation" technique into the standard software of modern TIMS instruments.

3. Corrections for effects such as mass fractionation or detector non-linearity are usually performed using known isotopic standards. Many laboratories use certified isotope reference materials provided by e.g. NIST/NBL or IRMM (e.g. IRMM-073/IRMM-074), but quite a large number also uses "consensus" type standards of natural uranium. The advantage of the better availability is often compromised by the lack of complete uncertainty propagation, leading to underestimated uncertainties and possibly biased results. There is an obvious need for more discussion and inter-laboratory knowledge exchange about guidelines for calculating uncertainties for isotope ratio measurements.

It is planned to organize measurement campaigns such as REIMEP 18 on a regular basis in order to re-assess the status of uranium isotope measurement capabilities, also to fulfil requirements for external quality control and to address upcoming measurement problems.

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# Remote Transmission of State of Health Information of Surveillance Systems

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***Abstract:***

*During the Liaison Subcommittee technical meeting organized in the frame of the Quadripartite Agreement between Argentina, Brazil, the ABACC, and the IAEA in Buenos Aires in September 2006, all the parties agreed that secure remote transmission and centralized collection and management of State of Health (SoH) information of running safeguards surveillance systems would enhance the efficiency of safeguards application. Taking into account that all the surveillance systems in Brazil and Argentina are under Joint Use between the ABACC and the IAEA, the SoH transmission opens the possibility of prompt reaction in case of occurrence of any important failure, despite of which agency is responsible for the routine maintenance of the surveillance system under consideration. The proximity of the ABACC headquarters from the facilities under control, and the joint use procedures agreed between both agencies, present an intrinsic advantage to mitigate the impact of re-verification measures due to the minimization of the period of potential loss of Continuity of Knowledge (CoK).*

*In this framework it was decided to start with a field trial of remote transmissions from SDIS surveillance system currently running at Angra II NPP in Brazil and from VIFM flow fuel monitor at Atucha I NPP in Argentina, giving the opportunity of one equipment from each organization be tested during this field trial.*

*Even though the IAEA has developed and applied remote transmission of SoH for several years, the ABACC has no experience in this field and was encouraged to develop its own system, using compatible design criteria with the IAEA. This decision obeys to the following factors: the ABACC and the IAEA must be able to provide equipment for Joint Use, and both agencies must be able to receive and manage the SoH information on-line and with no delay between partners. Besides, both organizations shall be technically able to have independent safeguards conclusions.*

*In this paper, the basic requirements and the resulting design of the SoH transmission system are described, emphasizing hardware, software and network security issues, data encryption and authentication. The first results of a laboratory trial are also shown. Finally, the next stages of the project, involving the design and development of the management system of collected information are depicted.*

**Keywords:** safeguards; surveillance; remote transmission; state of health

**Presentation**: Poster

# Illicit trafficking and border control - extended second line of defence

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***Abstract:***

*Recognising that we cannot achieve a 100% efficient global detection system that will catch every unauthorised move of nuclear and other radioactive materials, in strengthening our second line of defence there is a challenge to optimise the detection effort. The paper suggests focussing at development of in-field technology and definition of key nodal detection points, and discusses some elements of the defence in depth concept as extensions to its second line.*

**Keywords:** Illicit trafficking, border control, detection, defence in depth

## 1. Introduction

Coming into the new millennium, joint efforts by the Russian Federation and Finland at the European Union's North-Eastern border decreased the number of shipments with undeclared radioactivity from dozens to zero per year. The cases were typically contaminated scrap metal and orphan radiation sources (Hämäläinen [1]). The decrease in detections indicates the effectiveness of the Russian state system in particular and increased awareness among the shippers, which also renders the route unattractive for intentional smuggling.

Of course, this alone merely shifts the problem elsewhere; hence the need for focussing and extending our second line of defence.

## 2. Detection focus: reliable on-site tools at key nodal points

Now automatic radiation monitoring is used at international border crossing points: portal monitors for surveying the traffic and hand-held devices with search and dose rate functionality (Figures 1 and 2). The joint Finnish-Russian training programme for Customs staff is in a well established state.

### 2.1. Key nodal points

Recognising that we cannot reach 100% detection efficiency, in strengthening our second line of defence there is a need to optimise the effort. It can be achieved by focussing at development of technology and definition of the key nodal detection points. These points are not always physical locations, such as State and Union borders; they may be for example mass public events or related to important dates.

The points may be identified by risk assessment. It is then possible to concentrate stationary detection equipment, such as portal monitors, on the spots of high detection probability and to back them up by mobile units, the latter providing for enhanced deterrence through a random component and ad-hoc capacity (Figure 3).


**Figures 1 and 2.** Components of basic radiation monitoring at border crossing points: (above) stationary portal monitors for screening vehicle and passenger traffic, verification of alarms and detailed measurements by hand-held and movable systems, (below) provisions for in-depth inspections. Photos Reino Kaario/Customs.

### 2.2. In-field technology development

Another challenge is reliable detection and source identification. On CBRNE field missions one cannot afford to make serious false detections. Sophisticated mobile concepts require significant expertise in the data interpretation and decision making. Development of methods for non-experts will allow for a concept where the expert resources —instead of running in the field—concentrate on the assessment of the situation and in providing support. The next concept will be light-weight mobile non-expert nuclide identification systems, built on modern portable detectors, automatic measurement and analysis and near real time communication. Additionally, the vision is to share measurement data in a common database, available for expert analysis everywhere.


**Figure 3.** Sophisticated ON-line Nuclide Identification system, developed by the Security Technology Laboratory of STUK. A mobile laboratory is an efficient way to scan large areas on field missions to detect illicit radiation sources (Toivonen et al. [2]). Photo Petri Smolander/STUK.

## 3. Defence in depth

The zeroth line of defence in depth is deterrence: what takes place before the three lines of defence—prevention, detection and response—have to be actively applied. Its role is to discourage anyone from even attempting illicit actions on nuclear and other radioactive materials. This may be accomplished through elements of perception: a certain degree of visibility of security measures in place is required. Within the first line of defence, every significant batch of nuclear material and radiation source should be secured under a system of physical protection, accountancy and reporting. Should it fail, at least we should detect it and respond appropriately.

### 3.1. Knowledge bases and communication

In designing response to detection, timely acquisition of information on the seized material is crucial. Supplementary to direct measurements, one way to achieve it is through nuclear material databases and registries of radioactive sources directly and securely accessible to the authorities in a detection-response situation. Global databases enable occurrences of lost and found materials and sources to be tied together.

To some extent, the ‘winner’ of a nuclear terrorism incident will be the side with more effective media concept: capacity for timely and prompt public communication is of prime importance.

### 3.2. Attribution

The response in cases with criminal intent should not be limited to confiscation of the item and a legal process. Proactive response creates the basis of prevention: by applying nuclear forensics in attribution of the item we can find out its properties, origin, intended use, last legal owner, transport route (Wallenius et al. [3]). This ‘good circle’ strengthens prevention efforts through identifying sources of material not properly secured and mechanisms for illicit trafficking. Forensics and attribution also require global databases on materials and sources. Nuclear forensics are a powerful tool; the analyses, however, are

complex and time consuming and require highly specialised expertise. Here institutions such as the European Commission could have a crucial role: it would seem efficient to maintain joint nuclear forensics analysis services and establish procedures for members to submit items for analysis.

## 4. Cooperation between authorities

Due to the inherent global nature of illicit trafficking, cooperation between different authorities, nationally and internationally, is in a key role in combating it (Figures 4 and 5). A strong state system combined with global international cooperation will be effective also in fulfilment of the obligations set by international agreements such as the UN Security Council Resolution 1540, against proliferation of biological, chemical and nuclear weapons. The responsibilities undertaken by States Signatories to the Resolution include measures to combat illicit trafficking of sensitive materials.

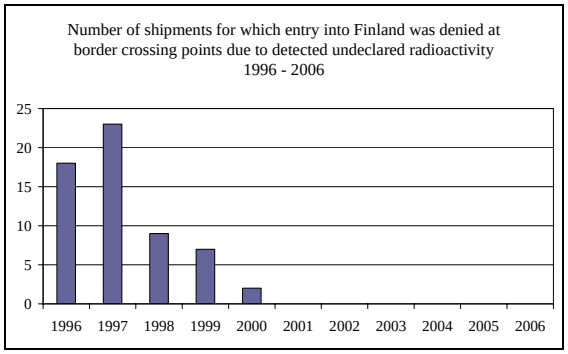


**Figure 4.** Joint efforts by authorities at the EU's North-Eastern border decreased radically the number of shipments with undeclared radioactivity, typically contaminated scrap metal and orphan sources. Efficient monitoring also helps deter intentional smuggling.

**Figure 5.** Cf-252 source hidden inside a structure of a transport pallet, 1993: underlining the importance of defence in depth and cross-border cooperation, the one nuclear smuggling case in Finland to date was detected and managed though cooperation between authorities across State borders, and consequently processed in the court of law. Photo Marko Hämäläinen/STUK.

## 5. Conclusions

In the optimisation of the second line of defence, development of detection may efficiently focus at reliable in-field technology and definition of key nodal detection points. Mobile units provide enhanced deterrence to border monitoring through an element of randomness and ad-hoc capacity. Methods for non-experts will allow for the expert resources to concentrate on assessment and support. Instead of limiting the response in cases with criminal intent to a legal process against the perpetrator, much may be gained by applying nuclear forensics in attribution of the seized item. Combined with maintenance of global databases on materials and sources this proactive response benefits future prevention efforts through identifying sources of material and mechanisms for illicit trafficking.

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# Automated System for Radionuclide Monitoring of Heat Carrier in the Primary Coolant Circuit of NPPs

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## *Abstract:*

*The automated system based on an HPGe detector with a 10% efficiency is intended for technological monitoring of the radionuclide specific activity of heat carriers in the primary coolant circuit of NPPs in the on-line mode. The measuring unit (MU) is a U-shaped glass tube embracing the cryostate cover on the level of the detector center which is placed in a lead shield for protection against the background of external radiation. The MU connection to technological systems is made with electromagnetic valves, whose position defines a system's operation mode. To ensure that there is a constant flow of the heat carrier, a bypass line is provided which connects the inlet pipeline with a system of organized through-flowing holes, specially made in the system to take the water for the analysis].*

*A special algorithm is developed to carry out the measuring procedure consisting of cyclic washing of the MU by water, taking measurements, and subtraction of the current value of background (residual) activity from the measured value of the heat carrier activity. The algorithm and software ensured a high reliability of the results at measuring the specific activity of the heat carrier at the presence of surface sorption of radionuclides in the MU.*

*The results of the measurements prove that the system is able to efficiently registrate the radionuclide specific activity in the total activity range in the heat carrier flow up to $7.3\times10^{6}$ Bq/l ($2\times10^{-5}$ Ki/l). The detection limit for the specific activity of the radionuclide $^{131}I$ is $1.83\times10^{3}$ Bq/l ($5\times10^{-8}$ Ki/l) at a measurement time of 600 s.*

*The automated system was proved to reliabe in conditions of NPP and can be used in other technological facilities for radionuclide-specific activity monitoring in different liquids and gaseous flows in the on-line mode.*

## 1. Introduction

The *γ*-radionuclide analysis of radioactive materials is one of the most informative methods of their non-destructive control, which is widely used in nuclear power engineering, the storage and processing of nuclear materials, environmental monitoring, and in safeguard [1]. The intensity and energy spectrum of the *γ*-radiation are unique characteristics of radionuclides which allow the composition and activity of radioactive materials to be determined.

Under the laboratory conditions, the radionuclide analysis of solid, free-flowing, and liquid materials is usually performed using standard spectrometric equipment – mainly based on HPGe detectors furnished with lead shields [2-4]. Providing the best precision of *γ*-radionuclide analysis, the mentioned laboratory equipment requires, at the same time, that there be provided sample intakes and appropriate sample preparation. In some applications, the portable instrumentation partially solves this task [5, 6]. However, by and large, the sample preparation (determining the speed of taking measurements) remains a tangible problem.

The general tendencies in automation of technological control processes are directed toward the creation of automatic systems for radionuclide analysis [7-9]. However, such equipment calls for non-standard solutions and should be closely linked to a particular application of a device.

The report presents the development results of the automated system for radionuclide monitoring of heat carriers of primary coolant circuits of NPPs. The developed system provides a measurement of the gamma-radiation nuclides spectra, automated procession, results display, data transfer, storage and documentation of radionuclide heat carrier flux content, and radionuclide activities.

## 2. Automated System

The automated system of the continuous monitoring of radionuclide contents in heat carriers of the primary coolant circuit (further – automated system) is intended to diagnose the state of the reactor active zone and to provide the limits and conditions of normal operation by the quantity of the leaky fuel rods. The control parameter is the activity of iodine radionuclides $^{131}$I – $^{135}$I, the indicators of the leakage of fuel rods shells - their regulation values are defined by the limits and conditions of safe operation of the power unit. Also, the activity of the radionuclides (corrosion products) - Cr-51, Fe-59, Mn-54, Mn-56, Nb-95, Co-60, Cs-134, Cs-137 etc. is to be controlled as it indicates the radioactive pollution of the technological mediums and equipment as well as emissions and discharges at NPP.

The circuit diagram of the automated system for radionuclide monitoring of heat carriers in the primary coolant circuit of NPP is shown in Fig.1. The system is comprised of:

- two automated spectrometers based on HPGe detectors [10] for two coolant circuits;
- data storage unit with software;
- automated system monitor for radiation control .

The automated system is connected to the server system of the central panel in the NPP radiation control service with a web-report system. Automated spectrometers and a data storage unit are placed in an unattended sampling room. The automated system monitor for radiation control is placed directly in the premises of the central panel of the NPP radiation control service.

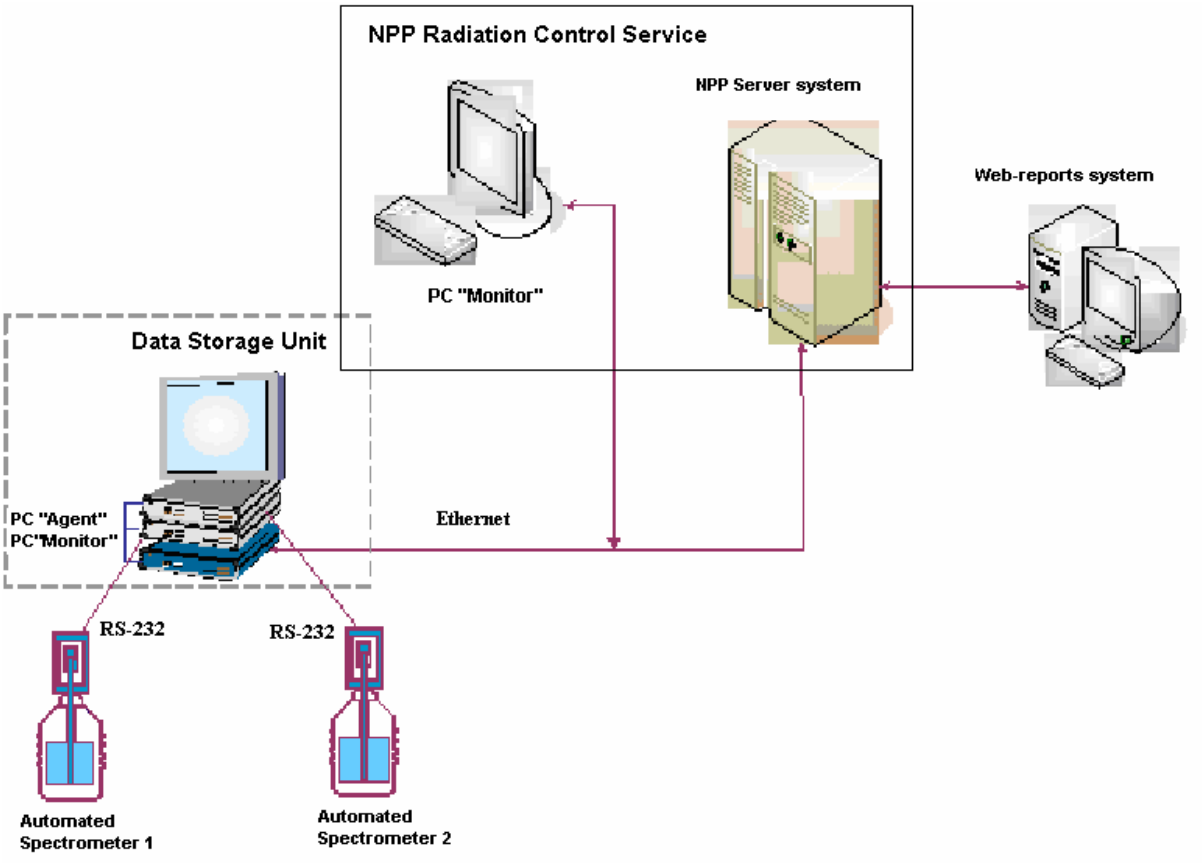


Fig.1. The system for radionuclide monitoring of heat carriers in the primary coolant circuit of NPP.

## 3. Automated HPGe Spectrometer

The automated spectrometer (fig.2) applies HPGe detector cooled by liquid nitrogen (BSI production), with an efficiency of 10% and standard values of energy resolution for the given efficiency [4]. Standard spectrometric electronics are used for amplifying and shaping the detector signals. The detector has a built-in gauge for control of the nitrogen level in the Dewar vessel, which sends a signal on a decrease in the level below the allowable limit as well as an automatic switch-off of the high voltage for feeding the detector when it exceeds the allowable temperature of the crystal. The time between refills of the detector by liquid nitrogen is not less than 14 days. If required, the spectrometer could be made on the basis of a HPGe detector with electro machine cooling [4].

Fig.2. The external view of the automated HPGe spectrometer

The measuring unit (MU) presents a U-shaped glass tube with anti-impact coating, embracing the cryostat cover at the level of the detector center in order to achieve the maximum efficiency of registration and, simultaneously, to make the maintenance and replacement more convenient. The detector and MU are placed in a lead shield to reduce the threshold of revealing radionuclides by passive protection of the detector against the background of the external radiation

For spectrometer calibration, a special calibration source (SCS) was made (fig.3). The SCS is a measuring unit filled by ion-exchanging resin impregnated with an Eu-152-containing solution, the specific activity of which is known, with its density of p=0.94. The primary certification of the SCS was carried out by the Center of Radiation Metrology of the Latvian National Metrological Center (the certificate of accreditation LATAK K-094), with the corresponding certificate issued.

Fig. 3. Special calibration source of the spectrometer

Figure 4 presents the SCS spectrum taken within 64000 s, on which the lines from 122 to 1408 keV are well distinguished. By the calibration spectrum, the curve of the real registration efficiency for radiation quanta was calculated, which was then used for calibration of the spectrometer.

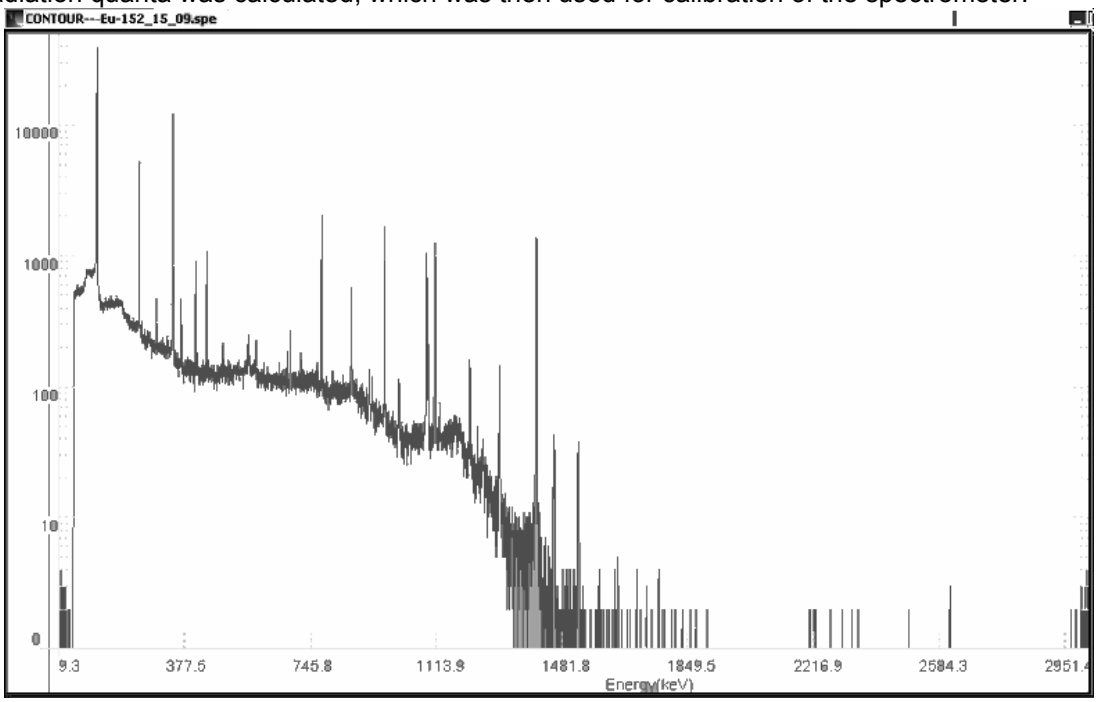


Fig.4. SCS spectrum

The offered design of the SCS simultaneously provides the following:
-simplicity of running metrological calibration tests, since all the preparation work consists of a routine replacement of the MU by a SCS and carrying out a measuring cycle using basic software.
- the capability to carry out a primary energy calibration of the spectrometric path, since the energy range of significant spectral peaks (122-1408 keV) is overlapping the entire measured energy range of the spectrometer.
- the capability to carry out a calibration of the detector's registration efficiency in the real geometry of measurement, without recalculation or introduction of any corrections, which considerably reduces the error of measurement.
-a long-term service life of the SCS without its replacement owing to a sufficiently large half-decay period of Eu-152 (13.54 years).

To realize the automated regime of measurements, the spectrometer has a special valve unit. It contains electronic flow meters for measuring the flowing liquid volume and a system of electromagnetic valves through which the spectrometer is connected to the technological systems Fig.5). In order for there to be a constant flow of the controlled medium, a bypass line is provided which connects the inlet pipeline with a system of organized through-flowing holes

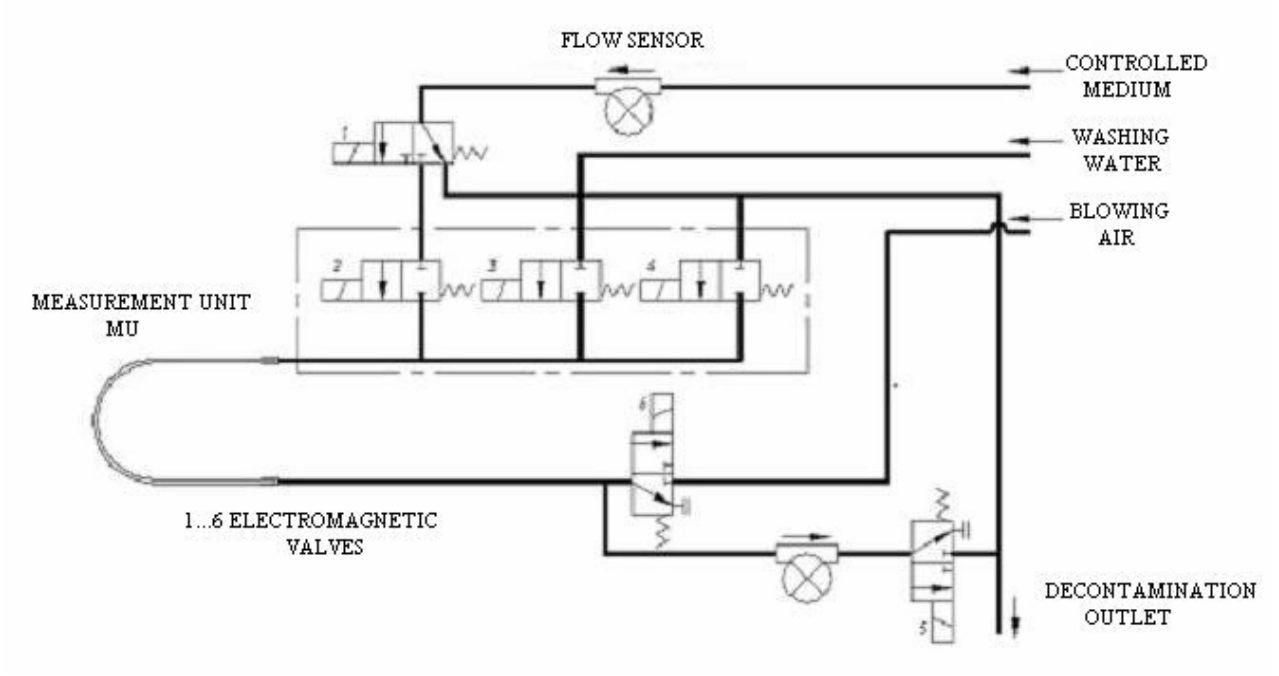


Fig.5. Hydraulic scheme of the valve unit

## 4. Data Storage Unit

The Data Storage Unit comprises 2 industrial computers – PC «Agent» and PC «Monitor» (See Fig.1).
PC «Agent» is a hard driver operating under OS Linux and providing control interfaces for two automated spectrometers, their electromagnetic valves, nitrogen level sensors in Dewar vessels, and flow liquid volume counters. PC «Agent» also performs the calculation of observable activities based on instrument spectra processing, archiving of the current values of observable activities, and the communication protocol for data and commands between PC "Agent", PC «Monitor» and the server system of the central panel of the NPP radiation control service, including diagnostics for network communications between sampling rooms and the central panel of the NPP radiation control service.

Two software packages, “Monitor” and “Service”, are installed in PC “Monitor” operating under OS Windows. The software package “Monitor” is intended for the system efficiency control, and monitors the radionuclide content of heat carriers and operation modes, the state of electromagnetic valves, etc.

The software package “Service” is intended for the adjustment of the configuration parameters of the system and auxiliary spectrometric tasks.

The personal computer, placed in the room of the central panel of the NPP radiation service, is a complete functional copy of PC «Monitor» in the set of the Data Storage Unit in the sampling room.

## 5. The Algorithm of the System Operation

The basic problem at development stage of the automated system was to ensure a long-term reliability of the results of measuring the activity of liquid or gaseous media in the presence of surface radionuclide sorption in the measuring unit.

To solve this problem, a special algorithm was designed, which was intended for the measuring procedure, which includes the cyclic washing of the MU with water or a special deactivating solution, taking measurements, and the subtraction of the current value of the background (residual) activity from the results of measuring the activity of the controlled medium. The switches between the modes “Measurements” and “Washing” are made automatically, in compliance with the set time characteristics of the modes or until the criteria of the washing quality are reached. The algorithm of the system’s operation in the modes “Measurements” and “Washing” is displayed on Fig.6.

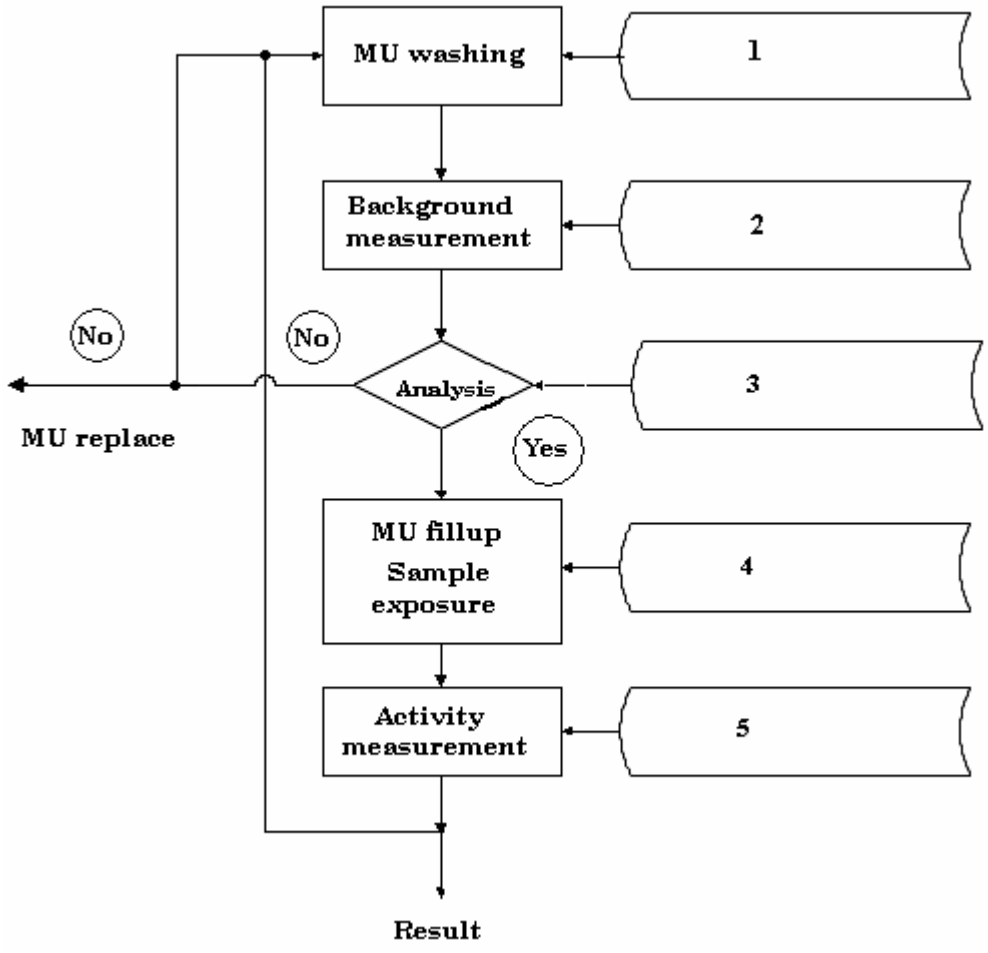


Fig.6. The algorithm of the system operation

According to the designed algorithm, the measuring procedure is divided into the following steps which are fulfilled in succession:

1. Washing of the MU.
2. Measuring of the residual (background) activity.
3. Estimation of the washing results (and, if necessary, repeated washing).
4. Filling of the MU with the medium to be controlled.
5. Measuring of the controlled medium activity.

The time of washing, measuring the background, filling the MU, and measuring the activity of the controlled medium as well as the cyclicity of taking the mentioned steps are set in the configuration software parameters of the spectrometer, with the possibility to operatively change all the parameters. The speed of the flowing medium is determined by the design of valves and controlled with electronic flow-meters which are connected to the data storage device.

If the background control gives a negative result after *n*-fold washing (point 3), a message is displayed regarding the necessity of replacing the measuring unit. At the arrival of the command to replace the measuring unit, an additional inlet valve for discharging the liquid opens, with air feeding for complete drying of the MU.

The measurements performed have justified the use of a U-shaped measuring unit, which ensures better technologies for MU making and its simpler replacement in the operational process, as well as a sufficient efficiency of the registration of the measured radiation for achieving the pre-defined level of the detection threshold. Employment of the measurement algorithm with subtraction of the current value of a discrete background as well as of the washing mode ensures, for a long time, the required value of the detection threshold without replacement of the MU.

Figure 7 displays a representative spectrum of the controlled medium (a) and a background spectrum after washing of the MU for 5 min (b) taken within 300 s.

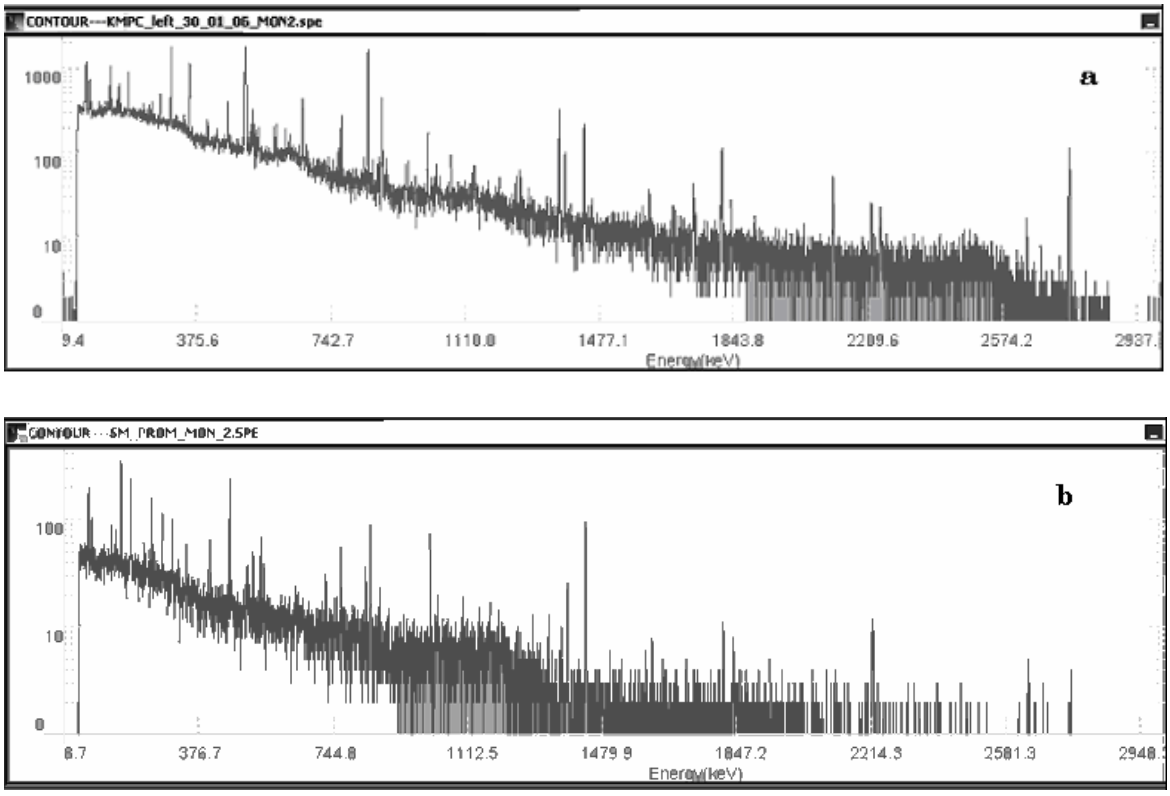


Fig.7.The spectra of the controlled medium (a) and a background (b) after washing the MU for 5 min.

Comparison of the spectra demonstrates that the efficiency of MU washing was not worse than $10^{-1}$. The efficiency of washing is defined as the ratio of the counts at the $^{131}I$ peak of the total absorption after washing to the corresponding counts at the peak when measuring the controlled medium.

From the results presented it is seen that, for the controlled group of iodine $^{131\text{-}135}I$, the MU washing provides the levels below the detection threshold in a measuring cycle with a single 5 min washing.

## 7. Conclusions

The developed system could operate in the automatic mode 24 hours per day; the time of the working mode setup does not exceed 0.5 hours. The time of setting each spectrum can vary from 1 to 100 000 s.

The range of the γ-quanta registered where the programmed support of the spectrometer is accomplished is (0.05-2.8) MeV. The absolute sensitivity to the flux of γ-quanta with an energy of 661 keV makes up $4.0\times10^{-3}$ pulses per quantum. The measurements have proved that the spectrometer provides efficient registration of radionuclide-specific activity in the total activity range of the measured liquid or gas flow up to $7.3\times10^{6}$ Bq/l ($2\times10^{-5}$ Ki/l). The detection limit for the specific activity of $^{131}I$ is $1.83\times10^{3}$ Bq/l ( $5\times10^{-8}$ Ki/l) at the measurement time of 600 s.

By its property of protection against the ambient medium (dust- and splash-proof), the spectrometer relates to the class which is not below IP33S [11]. The device keeps its characteristics in the operating temperature range + (5-40)ºC and humidity up to 95%.

The automated system has proved to be reliable in conditions of NPP and can be used in other technological facilities for radionuclide-specific activity monitoring in different liquid and gaseous flows in the on-line mode.

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# Swedish Support Programme (Swe SP) to IAEA Safeguards 20 year Anniversary

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***Abstract:***

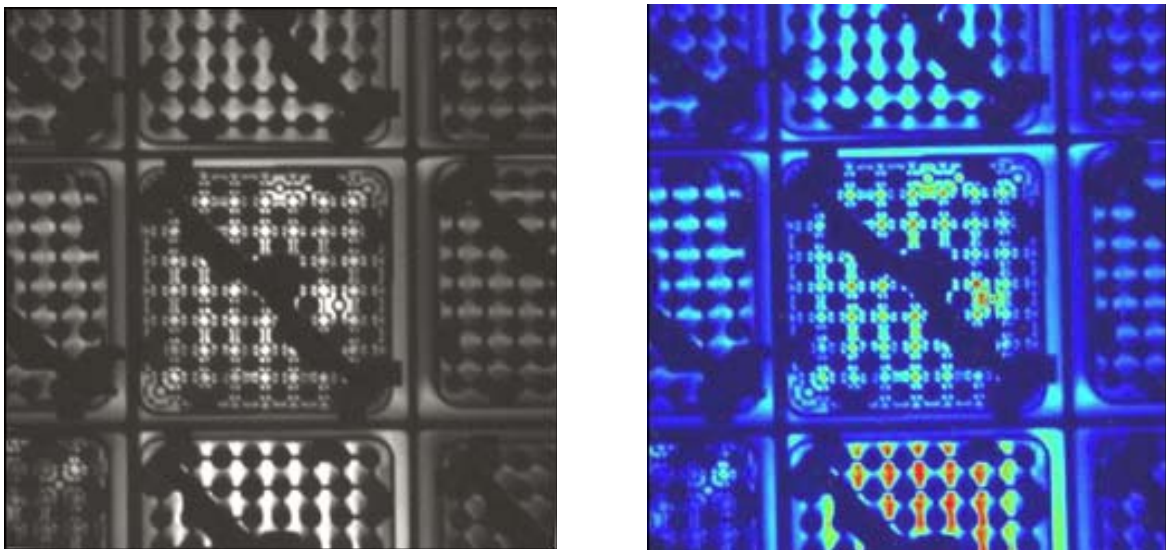
The Swedish Support Programme (SSP) was established in 1987 by the Swedish Parliament. It has given IAEA access to Swedish nuclear facilities for training of their inspectors and testing of verification equipment. Swedish experts have instructed at training courses, developed safeguards instrumentation and participated in expert groups. Several tasks are joint with other support programmes.

**Keywords: safeguards; support-programme**

## TRAINING IAEA

### ICVD&DCVD

There have been 27 courses on spent fuel verification since 1989. Training has been performed at the Ringhals and Oskarshamn sites. In addition some ICVD courses have been conducted in Finland and Spain. The focus of the courses has been on the use of the Cerekov Viewing Device, ICVD. In 2003 training on the SFAT (Spent Fuel Attribute Tester) was integrated in the course. This year the Digital CVD, DCVD, will also be integrated into the course. About 180 IAEA inspectors have been trained at Swedish facilities – totally 290. This is a joint task with the Canadian SP.



Spent BWR fuel viewed by a DCVD. The dark dots show the positions of the fuel-rods. There are five missing rods in this assembly. The picture to the right shows the same fuel in a colour coded image, red is most intense.

## Satellite Imagery Awareness

Swedish specialists on satellite imagery (from Metria Miljöanalys) give lectures for IAEA inspectors on the use of satellite date, the aim is to give the inspectors knowledge what can be done and what are the limitations.

Specialists from the IAEA Satellite Laboratory also come to Sweden for a field visit to selected facilities.

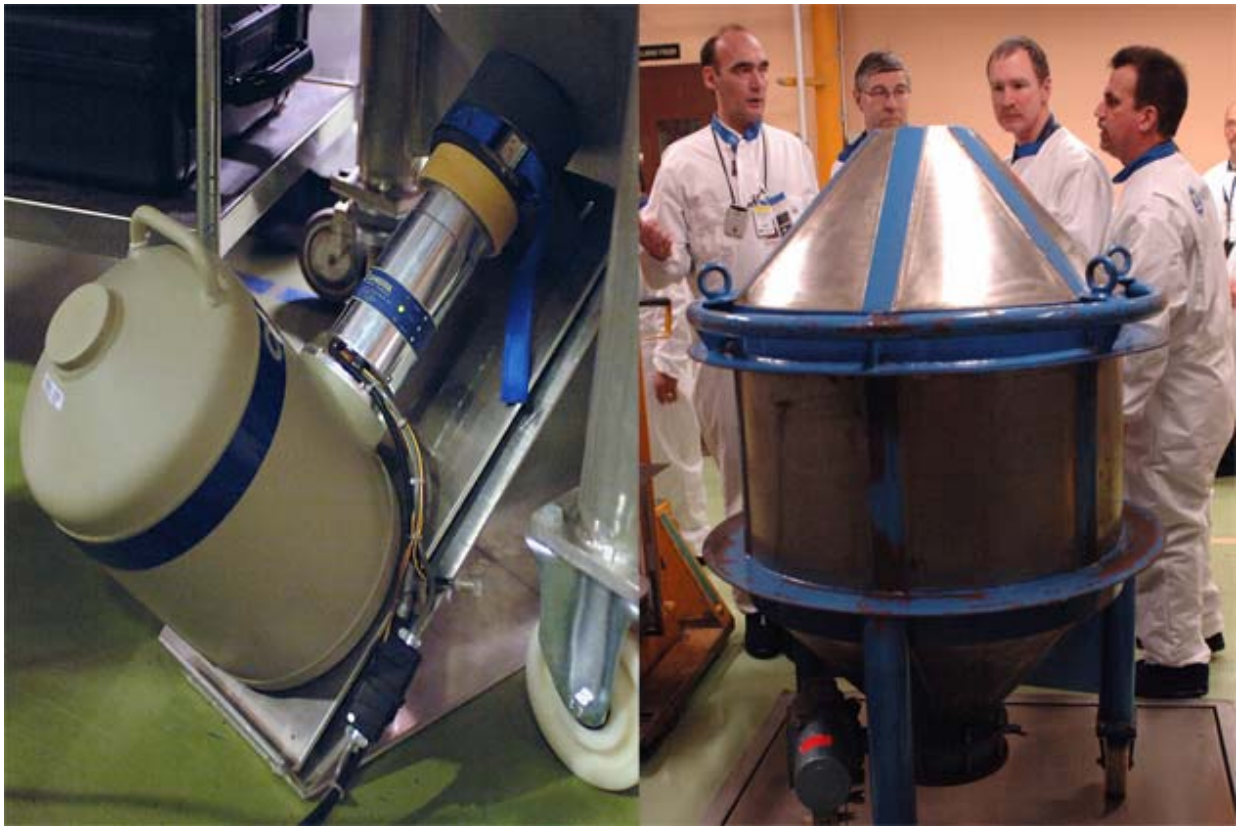
In 2007 specialists from the IAEA Satellite Laboratory will be trained in using the tool Metria CityModeler.

## Open Source information Collection

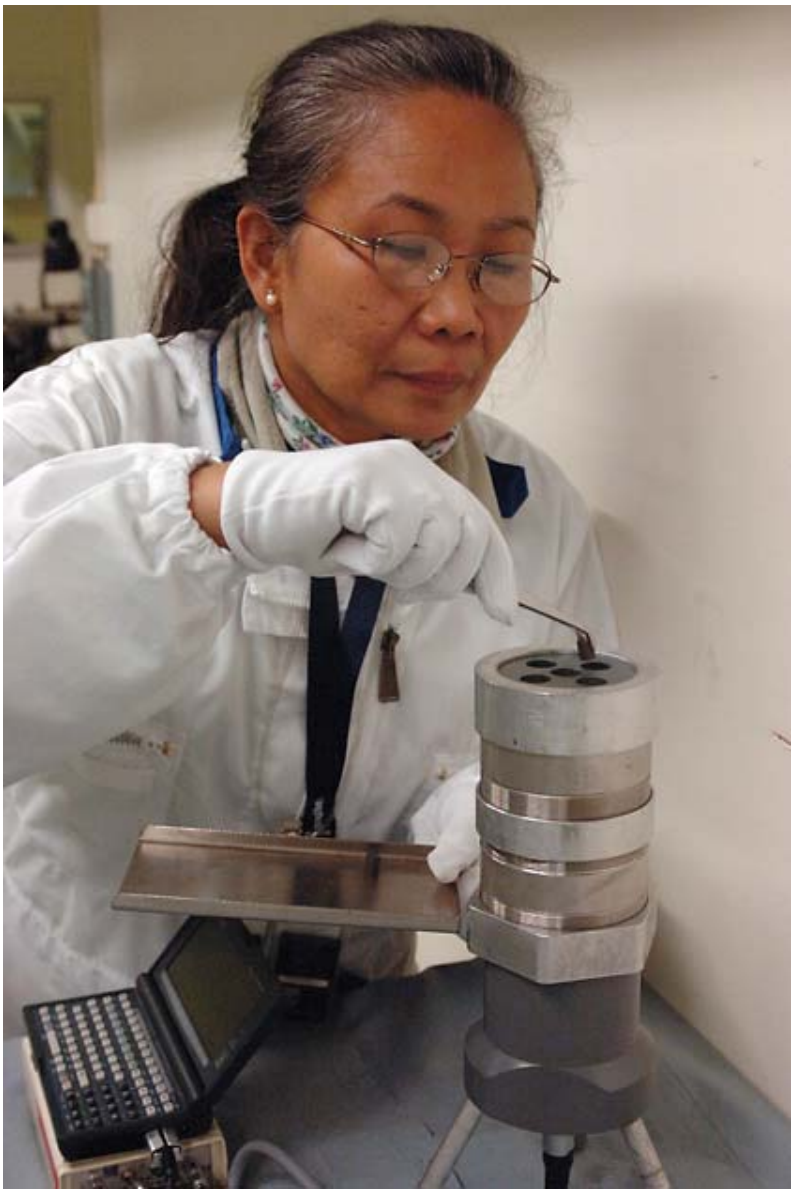
Swedish specialists on searching the Web conduct workshops for IAEA staff on how to use the internet in finding open source information.

## Bulk Handling Facilities

IAEA inspectors are trained on verification at a bulk handling facility at the Westinghouse Electric fuel fabrication plant in Västerås. The course was offered as an advanced training course from November 1977 (before the SSP was initiated) to November 1993. It was offered as a basic training course in 2001 and has been delivered every year since 2003. It is a challenge to have the course going while at the same time there is full production at the factory. As the fuel factory also has conversion from $UF_6$ to $UO_2$ powder this course is very valuable for the inspectors. A theoretical part is held in Vienna before arriving in Sweden.



Measuring uranium powder in a “hopper” with a germanium detector



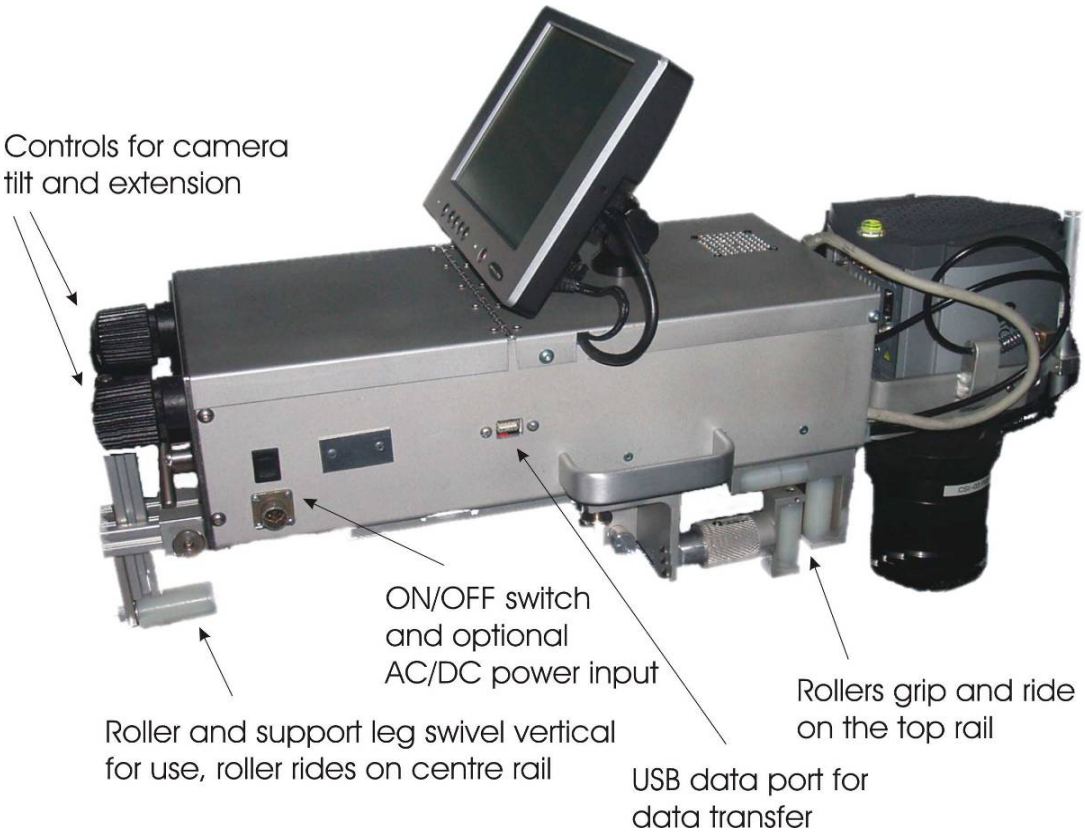
Measuring pellets for enrichment

## INSTRUMENT DEVELOPMENT

### A Digital Cerencov Viewing Device

The Swedish Support Programme has together with the Canadian SP developed a Digital Cerenkov Viewing Device, DCVD. The instrument has been approved by IAEA for inspection use. The DCVD offers the possibility of verifying long cooled, low burn-up spent nuclear fuel. Development continues with image analysis in search for missing and substituted rods within an assembly. The DCVD is also updated when more efficient sensors are available.

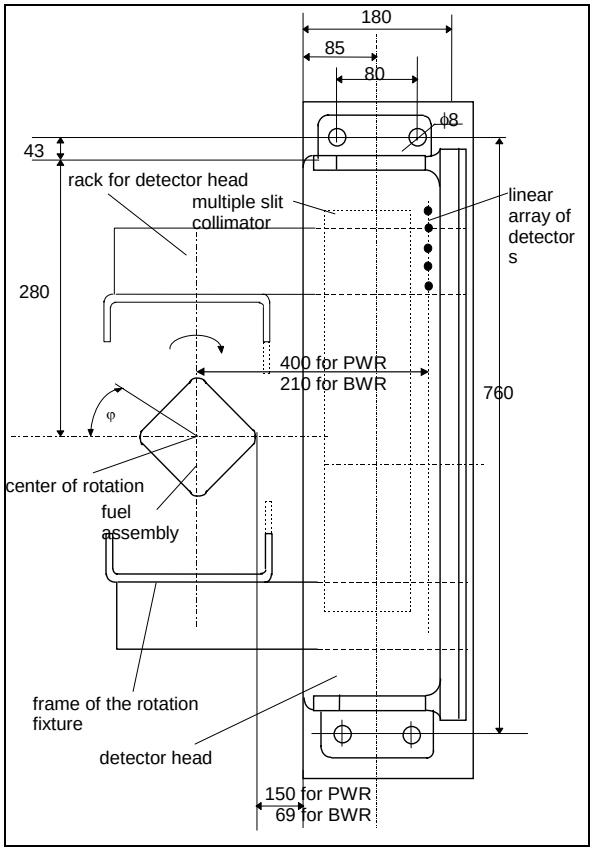
The screen on the laptop show what the UV-light detector sees in the pond.
This time a 105 mm lens is used, this lens is also used for the ICVD.



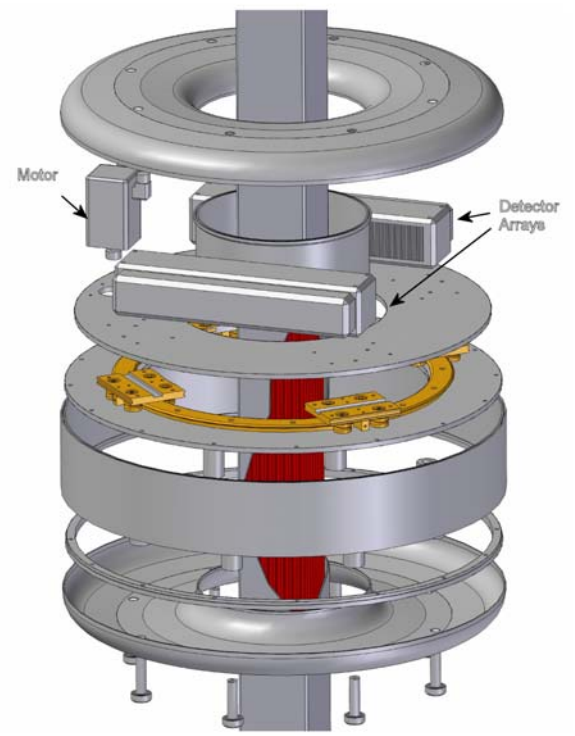

A never version of the DCVD. The instrument is built of Channel System Inc, Canada

## Tomographic Inspection Device

This is intended to be an instrument for detecting missing or substituted rods in a spent fuel assembly. Gamma radiation from the assembly is detected from several different directions. An algorithm can reconstruct the internal part of the assembly. This is a joint task with the Finnish and Hungarian SP:s. In 2007 the JRC Ispra will participate in the project.




Horisontal view of measurement principle for the prototype in 2002




Proposed design by Fin SP and Hun SP [1]

## EXPERT GROUPS

Swedish expert participate in different IAEA expert meetings. Swedish focus has been on Safeguards for the back end of the fuel-cycle. SweSP also tries to find Swedish experts to participate in meetings on “IAEA novel techniques”.

## BUDGET

The Swedish Support Program has on the average an annual budget of SEK 1 200 000 (appr. 130 00 euro). Most of the amount is spent on training.

Distribution in percent

| | |
|---|---|
| Administration | 3,0 |
| Training ICVD&DCVD | 32,5 |
| Training Bulk Handling | 30,8 |
| Training Satellite Awareness | 7,3 |
| Training Open Source | 8,7 |
| Development Tasks | 14,8 |
| Expert Groups | 2,9 |

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# Age determination of $^{233}U$

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***Abstract***

*The possibility to determine the age, i.e. the time since the last chemical separation, of $^{233}U$ was studied using the two fundamental different measurement techniques inductively coupled plasma mass spectrometry (ICP-MS) and gamma spectrometry. For the two materials analysed all measurement results were in agreement, i.e. consistent within the combined uncertainties. One of the materials was also measured using gamma spectrometry under field conditions. This measurement was also in agreement with the other results.*

**Keywords:** FMCT, age determination, $^{233}U$, gamma spectrometry, mass spectrometry, nuclear forensics

## 1. Introduction

Determination of different parameters in nuclear materials is important in many areas, e.g. nuclear safeguards, environmental studies and nuclear forensics (illicit trafficking). Materials covered by nuclear safeguards are basically plutonium, uranium (including $^{233}U$) and thorium [1]. One possible technique in a verification regime for a FMCT could be age determination, i.e. determination of the time since the last separation (e.g. chemical separation) of a material. Such a technique would make it possible to discriminate material produced before a FMCT taking action from material produced after the start of the treaty. In nuclear forensics, the age of a nuclear material is an important parameter in the characterisation of a seized material.

Age determination of plutonium and uranium can be done with different techniques. The application of different mass spectrometer systems for age determination of plutonium have been presented by several authors [2-4]. For age determination of uranium LaMont and Hall used thermal ionisation mass spectrometry (TIMS) [5], and Wallenius et al. [6] used TIMS and ICP-MS. Gamma spectrometric determination of plutonium age is for example implemented in commercial computer programmes like the FRAM software [7]. Age determination of high enriched uranium (HEU) using gamma spectrometry with intrinsic efficiency calibration was studied by Nguyen and Zsigrai [8]. All together these different techniques offer the possibility to determine the age of plutonium and HEU both for on-site inspections where high amounts of the material can be measured using non destructive analysis (gamma spectrometry), and of swipe samples, which has to be analysed in a specialised laboratory using more sensitive methods such as mass spectrometry.

In this work we demonstrate the possibility for age determination of $^{233}U$ using both gamma spectrometry and ICP-MS.

## 2. Theory

The determination of the age of $^{233}U$ was based on the mother/daughter relationship between $^{229}Th$ and $^{233}U$. The relationship is described by the following equation

$$t = \frac{1}{\lambda_1 - \lambda_2} \cdot \ln\left[1 - \left(\frac{\lambda_2}{\lambda_1} - 1\right)\frac{N_2}{N_1}\right] \quad (1)$$

where $t$ is the age of the material, $\lambda_1$ and $\lambda_2$ are the decay constants of the mother and the daughter nuclide and $N_1$ and $N_2$ are the amounts of the mother and daughter nuclide at the time $t$. The half-lives used in the calculations were obtained from the Evaluated Nuclear Structure Data File (ENSDF) [9].

## 3. Experimental

All analysis were done on the isotopic reference material IRMM-040a in an acid solution (IRMM, Geel, Belgium), and on a $^{233}U_3O_8$ material (AEA Technology, England) (material 2). Isotope dilution mass spectrometric measurements (ID-MS) of the $^{229}Th/^{233}U$ ratio were done using an Element2 sector field ICP-MS (Thermo-Finnigan, Bremen, Germany). Prior to the determinations, uranium and thorium were separated by means of solid phase extraction using TEVA and UTEVA resins (Eichrom, Darien IL, USA). The ICP-MS instrument was equipped with a semi-demountable Fassel quartz torch and a CD-2 guard electrode. Sample introduction was performed using a conical nebuliser and a cyclonic spray chamber. Self-aspiration was used resulting in a sample flow rate of about 0.2 ml $min^{-1}$. Measurements were done using low resolution ($m/\Delta m$=300).

Gamma spectrometric measurements were done using two systems. Laboratory measurements were done with a coaxial high-purity germanium (HPGe) low-energy photon detector having a resolution of 0.65 keV at 60 keV (LOAX, EG&G Ortec, USA). On-site measurements of material 2 was performed with a coaxial HPGe detector with a 110% relative efficiency and a resolution of 1.9 keV at 1332 keV (EG&G Ortec, USA). A relative response function was established using gamma rays from $^{233}U$ in the range 118 to 365 keV. $^{229}Th$ was thereafter evaluated using calculated responses from the response function and the $^{229}Th/^{233}U$ amount ratio was calculated. Tab 1 gives gamma rays from $^{233}U$ and $^{229}Th$ used in the measurements.

| $E_\gamma$ $^{233}U$ (keV) | $E_\gamma$ $^{229}Th$ (keV) |
|---|---|
| 118.97 | 136.99 |
| 120.82 | 142.96 |
| 135.36 | 154.34 |
| 146.35 | 156.41 |
| 164.52 | 179.76 |
| 208.17 | 183.93 |
| 245.35 | 193.51 |
| 248.73 | 204.69 |
| 291.35 | 210.85 |
| 317.16 | |
| 320.54 | |
| 336.61 | |
| 365.79 | |

**Tab. 1 Gamma ray energies of $^{233}U$ and $^{229}Th$ used in the age determination.**

Combined uncertainty calculations were done on all measurement results according to ISO/GUM [10] using the software GUM Workbench [11]. All reported uncertainties are expanded uncertainties with a coverage factor of two ($k$=2), unless otherwise stated.

## 4. Results and discussion

Fig 1 shows the fit of the response function to data points from $^{233}$U and normalised data for $^{229}$Th for a measurement of IRMM-040a using the LOAX detector. The uncertainty of each data point ($k$=1) in the figure is a combined uncertainty including uncertainties coming from counting statistics, the response function, branching ratio and half life.

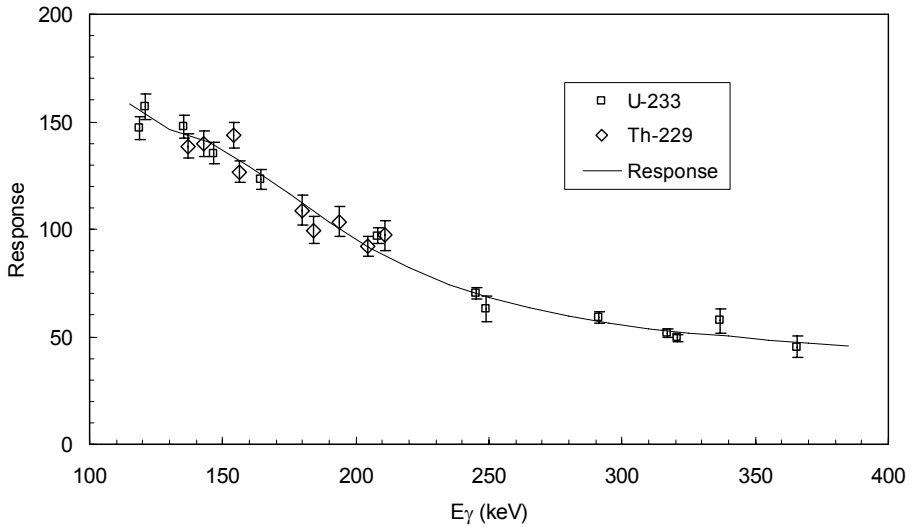


**Fig. 1 Data points from $^{233}$U, $^{229}$Th , and the fit of the response function for the measurement of IRMM-040a using the LOAX detector.**

Tab 2 shows the result from the age determinations of the two different materials using gamma spectrometry and ICP-MS. As can be seen the measurement results for each material using the two methods corresponds well within their combined uncertainties.

| Measurement | Age of IRMM-040a | Age of Material 2 |
|---|---|---|
| Gamma spectrometry (LOAX) | (29.8±1.6) y | (41.2±2.6) y |
| Gamma spectrometry (TF2) | *not measured* | (42.7±3.4) y |
| ICP-MS | (29.9±0.3) y | (42.9±0.4) y |

**Tab. 2 Results of the age determinations using the $^{233}$U/$^{229}$Th ratio. (Reference date: 2007-05-01).**

There are no reference dates available for the materials used in this work. However, confidence in the results can be obtained using different independent methods, i.e. methods based on completely different measurement approaches. The methods used in this work: gamma spectrometry and ICP-MS differs on a fundamental basis. For example, in mass spectrometry the analyte ion is detected (e.g. the $^{233}U^{+}$-ion,) whereas in gamma spectrometry the gamma ray photons emitted in the radioactive decay of e.g. $^{233}$U is detected, i.e. different physical processes are used in the different measurement approaches.

## 5. Conclusions

In this paper we have presented two methods for age determination of $^{233}$U: Gamma spectrometry and mass spectrometry using sector field ICP-MS. Both methods resulted, within combined uncertainties, in the same age for the two measured materials. The use of fundamentally different measurement techniques resulting, within the combined uncertainties, in the same result serves a validation of the different techniques and giving confidence in a measurement result.

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Wyhlen; Germany

# Twenty years of HOLDUP: A way to enrich the safeguard concept.

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ESARDA 29th Annual Meeting, Symposium on Safeguards and Nuclear Material Management, Aix-en-Provence (France), May 22-24, 2007

***Abstract***

The holdup measurements have been carried out for twenty years in the Valduc facility. This experience can be applied directly to the safeguard concept.

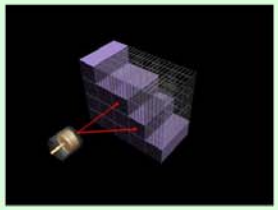
The Holdup measurements in large objects (glove-box, tank) consist in determining plutonium's activity by the infinity energy method. The structure is virtually cut out in small individually measured volumes. A solid angle correction is applied on each measurement to determine a global activity.

This simple and robust method allows a rapid determination of the activity and requires a portable HPGe detector with a resolution less than 750 eV. The matter's quantity calculation is based upon the emission of 239Pu and 241Pu. The calibration is carried out with a point source or a simple modeling.

This poster describe the measurement technical and specifies its performances.

## HoldUp

- Cutting the glovebox in small volumes
- Use infinite energy for each volume
- Adjustement of the Pu mass with a calculation of solid angle



## Detector

- BEGe 3800 mm² or type P 40 %
- Resolution 129 KeV : 640 eV at 750 eV
- ISOCS characterization
- Portable cryostats

## Measurement

- No collimation
- Energy calibration < 200 eV/canal
- Aiming in the axis of the détector

## Performs

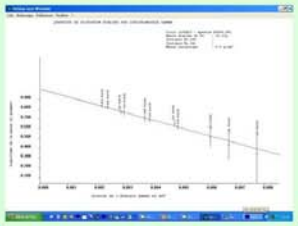
- Detection limits 0.1 g Pu (counting time 15 hours)
- Counting time : few minutes to a night
- Fast calculated

## Some results – during all these years

- 36 Glove boxes dismantling
- 450 g of Pu measured with HoldUp
- 511 g of Pu measured in passive neutron during dismantling

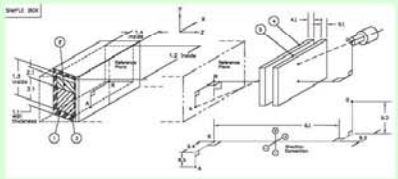
## Infinite Energy

- Straight (1/Energie) = Log (Mass Pu)
- Use 17 absorption peaks
- Isotopics $^{239}Pu$ & $^{241}Pu$
- Ratio 129/413 KeV for détermination of absorption



## Efficiency

- Modelling the objects filled of contaminated air



## Objects

# Comparison of calculated and measured efficiencies for a neutron slab counter for a variety of sealed sources of different energy spectra

**Stephen Croft[1], Eloisa Alvarez[2], Patrick Chard[2] and Ian Hutchinson[2]**

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***Abstract:***

*The ESARDA Non-Destructive Assay (NDA) Working Group (WG) recently completed an intercomparison exercise designed to evaluate the reliability and accuracy of the Monte Carlo method and associated evaluated nuclear data for the calculation of neutron detection efficiencies. The exercise was referred to as the 'simple case' because the detector chosen for study was a neutron slab detector which conceptually, at least, is one of the most basic NDA instruments. A number of groups modelled a number of configurations involving various arrangements of moderator sheets and cadmium attached to the slab. A range of Monte Carlo codes, physics treatments and candidate nuclear data sets were used. All of the configurations involved a point-like neutron emitter of $^{252}$Cf at 500mm. In addition to comparing the calculation results from the different groups that took part, to reveal how user choices might influence the outcome, a comparison was also made to carefully performed and evaluated experimental values.*

*In this work we have extended the 'simple case' study by including experimental results for $^{252}$Cf taken over the range 200mm to 1000mm. In addition data were gathered at 500mm and 1000mm using certified ($\alpha$,n) sources of Am/Li, Am/F, Am/B and Am/Be. These extra measurements allow the Monte Carlo study to be extended and in particular the sensitivity to the representation of the ($\alpha$,n) spectrum to be assessed.*

*We also present additional Monte Carlo characterisation data for the N50 slabs, showing the angular response variability.*

**Keywords:** MCNP; neutron detection; N50; slab; simple case; neutron spectra

## 1. Introduction

The ESARDA Non-Destructive Assay (NDA) Working Group (WG) has organised several intercomparison exercises, aimed at establishing the performance of NDA techniques currently employed in safeguards. The 'simple case' exercise [1] was designed to evaluate certain aspects pertaining to the accuracy and reliability of Monte Carlo modelling for calculating neutron

detection efficiencies since Monte Carlo modelling is becoming increasingly widespread as a tool in the calibration of neutron coincidence counting systems.

In the so called 'simple case' exercise a number of groups modelled different geometries involving a point-like $^{252}$Cf source of certified emission rate at 500mm distance from a model N50 neutron slab monitor [2,3] with various thicknesses of moderator and or Cd in between. *A range of Monte Carlo codes, physics treatments and candidate nuclear data sets were used.* The calculated results from the different groups were compared with carefully performed and evaluated experimental values.

In this work we present experimental results taken over the range 200 to 1000mm for the $^{252}$Cf source. In addition data were gathered at 500mm and 1000mm using certified ($\alpha$,n) sources of Am/Li, Am/F, Am/B and Am/Be. We have also included an extra configuration with 1mm of Cd on the front face for some of the runs.

MCNPX$^{TM}$ v2.5 [4,5] and mostly ENDF-VI.8 cross-section data was used to extend the Monte Carlo study to the new measurements. MCNPX$^{TM}$ is a superset of MCNP4C, one of the codes the participants in the 'simple case' used. The cross-section data release version is more recent that the ones used during the intercomparison exercise. Note that for the polyethylene ENDF-VI.6 cross-section was used. Different neutron source spectra were used in the simulations to represent the $^{252}$Cf source and various ($\alpha$,n) sources as part of a sensitivity analysis.

## 2. Measurements using $^{252}$Cf source

The $^{252}$Cf source (Cf-8761NC) was placed at different distances from the front face of the bare N50 slab, located on the centreline of the detector array, and at the midpoint of the active length of the detectors. The slab monitor was mounted 1.645 metres from the ground, in order to minimise the contribution to the count rate from neutrons which have been in – scattered from the environment, principally the concrete floor of the laboratory. A Cd sheet covered the floor area near the detectors, to provide further protection against re – entrant slow neutrons. The experimental set-up was originally designed with the 'simple case' intercomparison in mind and for that a source-to-slab separation of 500mm was used. The Cd sheet had dimensions of about 1.2m wide x 1.5m in the direction of the source-to-slab. It was not enlarged when the extra measurements at greater distances were performed.

The detection efficiency determined as a function of separation is summarised in Figure 1. The experimental data were corrected for room return neutrons using a combination of the shadow cone technique and an analysis with distance in which the constant term was ascribed to the sea of neutrons created by scatting off the floor, walls and ceiling. A detailed uncertainty study was performed in order to assess the overall random uncertainty associated with these results and included contributions from: counting statistics; room return; ambient background subtraction; decay corrected source emissions strength; reproducibility of positioning etc. The overall standard deviations are shown as error bars in Figure 1.

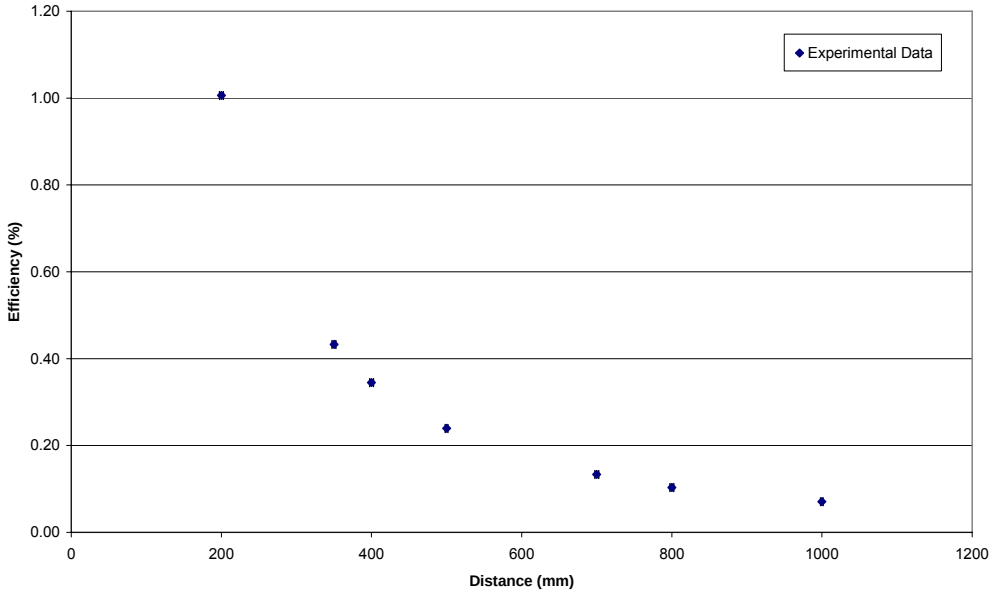


**Figure 1:** $^{252}$Cf N50 slab experimental efficiency vs distance.

## 3. Measurements using Am/Li/B/Be/F sources

Measurements with the ($\alpha$,n) sources were taken at 500mm and 1000mm from the front face of the N50 slab in a similar geometry to the previously presented $^{252}$Cf measurements. An extra configuration with 1mm of Cd on the front face was included for some cases. The sealed sources used are summarised in Table 1. They were supplied by Amersham International; a more detailed description can be found elsewhere [6]. Table 1 shows the anisotropy correction factors applied to each of the sources, these correction factors account for the fact that the emission through the cylindrical surfaces is greater than through the ends and therefore a correction factor has to be applied to the 4π source strength that appears in the source certificate. The fluence rate F(r, J) at a point, P, a distance r from the centre, O, of the active material within the sealed source measured along OP which subtends a plane angle , J, with repect to the cylindrical axis OT through the top, that is the external weld end of the source is given by:

$$F(r, J) = (S_n/4.p.r^2) \,.\, F_I(J)$$

where $F_I(J)$ is the anisotropy factor.

| | | Anisotropy Factors at 90 deg: | | | |
|---|---|---|---|---|---|
| Source | Capsule | $F_I(\pi/2)$ | Std. Dev. | RSD (%) | Eff. CF |
| Am/Li-5831 | X.3 | 1.033 | 0.01 | 0.968054 | 0.968054 |
| Am/F-5828 | X.2 | 1.016 | 0.005 | 0.492126 | 0.984252 |
| Cf-8761NC | X.1 | 1.006 | 0.002 | 0.198807 | 0.994036 |
| Am/B-3007 | X.2 | 1.015 | 0.005 | 0.492611 | 0.985222 |
| Am/Be-326 | X.2 | 1.013 | 0.005 | 0.493583 | 0.987167 |

**Table 1:** Source Details

The X.2 assembly has an outer diameter of 17.4mm, is approximately 19.2mm long and has a combined effective thickness of (2.94±0.1)mm of stainless steel on the base. The source pellet is

(14.00±0.03)mm in diameter and nominally (11.6±0.1)mm long. The wall thickness is approximately 1.6mm. The X.3 assembly is slightly larger. It has an outer diameter of 22.4mm, is 31.1mm long and has a combined effective thickness of about (5.06±0.1)mm of stainless steel on the base. The wall thickness is approximately 2.4mm. The source pellet is (17.50±0.03)mm in diameter.

The detection efficiency was measured for each of the sources and a detailed uncertainty study, similar to that undertaken previously [1], was performed in order to asses the overall random uncertainty associated with these results including uncertainties originating from the engineering tolerances in the experiment and the detector manufacture. The efficiencies, corrected for room scatter, are presented in Table 2. The standard deviation quoted is the quadrature sum of all significant contributions including: counting statistics; room return; ambient background subtraction; decay corrected source emissions strength; the anisotropy correction factor for the source; reproducibility of positioning etc. Not included, but which affect the measured to calculated comparison, are dependences on the slab construction (HDPE density, dimensional inaccuracies, $^{3}$He partial pressure, effective active length) and operational choices (e.g. where on the HV plateau the voltage was set). These matters have been quantified however and are not expected to amount to more than about ±0.7% at 1-σ.

| Source | Cd | Distance (mm) | Room scatter corrected $\varepsilon$ (%) | Overall $\sigma_\varepsilon$ (%) |
|---|---|---|---|---|
| Am/Li-5831 | N | 1000 | 0.0808 | 0.0021 |
| Am/F-5828 | N | 1000 | 0.0753 | 0.0018 |
| Cf-8761NC | N | 1000 | 0.0695 | 0.0015 |
| Am/B-3007 | Y | 1000 | 0.0580 | 0.0008 |
| Am/B-3007 | N | 1000 | 0.0594 | 0.0006 |
| Am/Be-326 | N | 1000 | 0.0502 | 0.0014 |
| Am/Li-5831 | N | 500 | 0.2788 | 0.0044 |
| Am/F-5828 | N | 500 | 0.2589 | 0.0036 |
| Cf-8761NC | N | 500 | 0.2382 | 0.0033 |
| Cf-8761NC | Y | 500 | 0.2406 | 0.0027 |
| Am/B-3007 | N | 500 | 0.1936 | 0.0025 |
| Am/Be-326 | N | 500 | 0.1639 | 0.0034 |

**Table 2:** Measured efficiencies ($\varepsilon \pm \sigma_\varepsilon$) at 500mm and 1000mm for $^{252}$Cf and ($\alpha$,n) sources: Am/Li/B/Be/F.

## 4. Monte Carlo Simulations and Results

In this work MCNPX$^{TM}$ v2.5 and the ENDF-VI cross-section data has been used to simulate the experimental conditions presented above. The N50 neutron slab was modelled following the geometric assumptions previously established for the 'simple case' intercomparison exercise and all the different sources have been considered point-like sources and the source capsules have not been included in the models. This is not a limitation since the experimental values were corrected for source anisotropy and the emission spectra typically include source perturbation effects. Different spectra have been considered to simulate the $^{252}$Cf :

1. A Watt fission energy spectrum that is built into MCNPX$^{TM}$ with the shape:

$$p(E) = C\exp(-E/a)\sinh(bE)^{1/2}$$

where a = 1.025 MeV,
b = 2.926 MeV$^{-1}$

2. Maxwellian ISO standard spectrum [7], defined in the range 100 keV – 10 MeV.

3. A modified Maxwellian spectrum according to Grundl and Eisenhauer [8].

4. Watt spectrum fit according to Froehner et al. [9].

The deviation from the experimental results vs the distance can be seen in Figure 2 for the various neutron energy spectra described above. The error bars are based solely in the experimental values (the Monte Carlo calculations being run for sufficient time to achieve less than 1% relative statistical standard deviation) and are calculated for the percentage difference from the calculated (C) to the measured (M) efficiencies as $100.C.\sigma_M/M^2$, where $\sigma_M$ is the estimated standard deviation in the measured value.

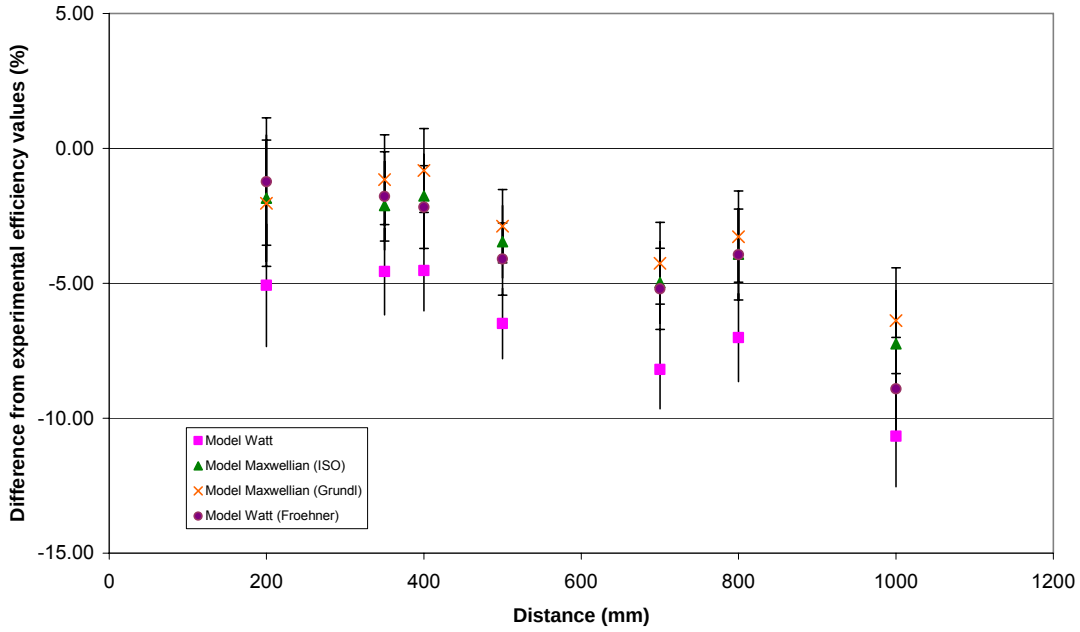


**Figure 2:** $^{252}$Cf N50 efficiency vs distance.

Figure 2 shows that the agreement between the modelled and experimental results is less than 5% for separations less than 500mm. The difference between experimental and modelled results increases

with the distance, reaching up to 13% difference at 1000mm distance in the worst case. It can be observed as well that the MCNPX™ models using the Watt spectrum produced consistently poorer agreement across the full range of distances than the other spectra modelled in MCNPX™, which is consistent with previous findings [1]. The modified Watt spectrum of Froehner [8] came closest to the experimental values.

The experimental data was corrected for room return neutrons using a combination of the shadow cone technique and an analysis with distance in which the constant term was ascribed to the sea of neutrons created by scattering off the floor walls and ceiling. The room measured approximately 8.6m by 10.1m with only a light weight high ceiling. The trend apparent in the difference between the experimental and calculated efficiency with distance is apparently much larger than the ascribed uncertainty in the room correction.

The main shadow cone technique we used involved placing a cone of high density polyethylene with a steel base (this was a module which was designed for Van der Graaff experiments) between the source and the slab to block the direct shine path leaving only the room return contribution. The effect at 500mm was modest (estimated at about 0.6% for $^{252}$Cf and 1.2% for Am/Be). A 50% uncertainty was assigned to allow for the fact that the shadow shield was conical and therefore did not provide a perfect geometrical match to the slab.

The room return correction to the apparent efficiency at distances other than 500mm was assumed to be a constant independent of separation. In other words the room scatter was assumed to create a uniform flux of neutrons in the region where the measurements are conducted. Detailed confirmation of this assumption was not however performed because of the emphasis of the measurements being on the 500mm separation geometries. A 50% uncertainty in the room return correction was propagated. The systematic trend in the data may indicate that the importance of room return is greater than predicted by this simple treatment. Non-the-less the experimental results at 500mm and less, where the room return is fractionally less important, are believed to be valid and unbiased within the uncertainty analysis presented.

Different candidate spectra were also used in the modelling of the Am/Li/Be/B/F sources. The neutron energy spectra for the Am/Li and Am/F sources was adopted from Owen et al,[ 10]. From Marsh et al [11] high resolution neutron measurements at the NPL, two spectra for the Am/Be and one for the Am/B source have been used in the Monte Carlo modelling. The ISO standard spectra have been used for the Am/Be/B sources, a further spectrum for the Am/Be source was obtained from Kluge et al [12]. From Jacobs and Liskien [13], the energy spectra of neutrons produced by 5.5 MeV α-particles on a thick B target was tried.

The MCNPX™ modelled efficiencies vs the mean spectrum energy for the different sources can be seen in Figure 3. As expected the neutron detection efficiency decreases with the spectrum mean energy. The neutron efficiency differences from the modelled to the experimental results are presented in Table 3. The deviations have been calculated according to:

$$100\cdot(C/M-1) \text{ +/- } 100\cdot C\cdot\sigma_M/M^2$$

where C corresponds to the Monte Carlo calculated efficiencies, M to the measured ones and $\sigma_M$ is the standard deviation in the measured value. Once again the sampling precision in the Monte Carlo runs was small in comparison.

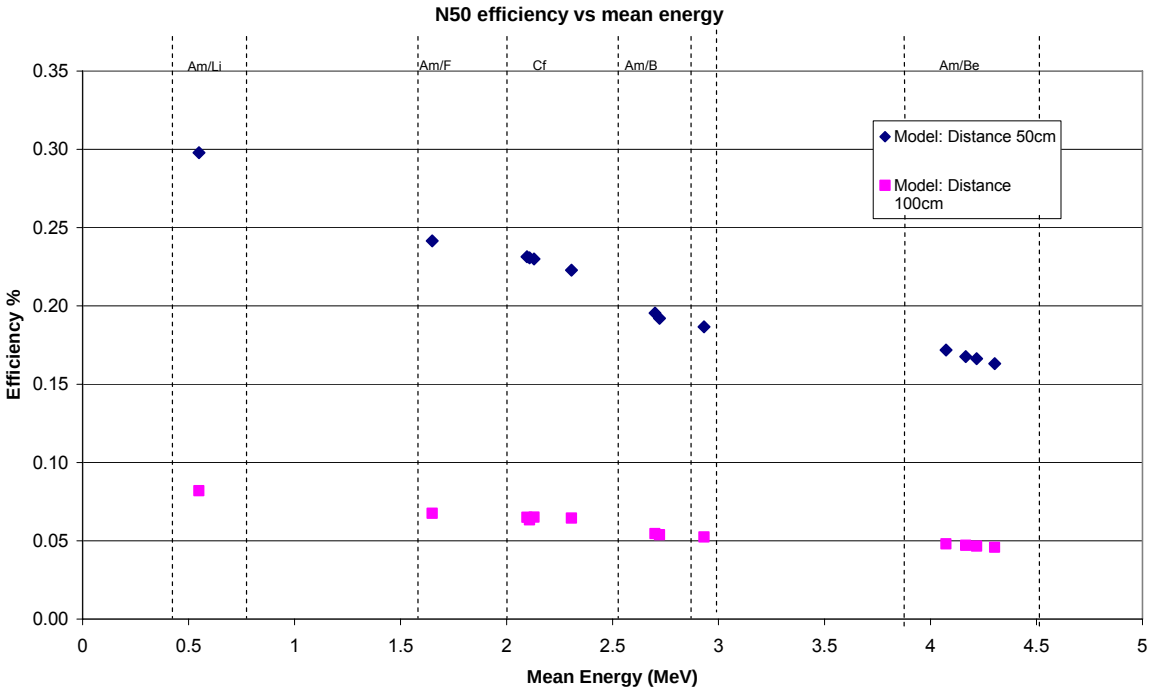


**Figure 3:** MCNPX$^{TM}$ N50 slab neutron efficiencies vs spectrum mean energy.

| Source Spectrum | Mean Energy (MeV) | Difference at 500cm (%) | Uncertainty in the % Difference at 500mm | Difference at 1000mm (%) | Uncertainty in the % Difference at 1000mm |
|---|---|---|---|---|---|
| Am/Li (B'ham) | 0.55 | 6.84 | 1.70 | 1.45 | 2.62 |
| Am/F (B'ham) | 1.65 | -6.72 | 1.28 | -10.34 | 2.09 |
| Am/B (ISO) | 2.72 | -0.80 | 1.30 | -9.34 | 0.88 |
| Am/B (NPL) | 2.70 | 0.90 | 1.33 | -8.06 | 0.89 |
| 5.5MeV α's on a B target (Jacobs & Liskien) | 2.93 | -3.56 | 1.67 | -11.68 | 0.85 |
| Am/Be ( 'Large' NPL) | 4.07 | 4.80 | 2.16 | -4.52 | 2.57 |
| Am/Be ( 'Small' NPL) | 4.30 | -0.48 | 2.05 | -8.78 | 2.46 |
| Am/Be (Kluge & Weise) | 4.17 | 2.25 | 2.10 | -6.29 | 2.53 |
| Am/Be ( ISO) | 4.22 | 1.52 | 2.09 | -7.17 | 2.50 |
| $^{252}$Cf Watt | 2.31 | -6.49 | 1.30 | -7.24 | 1.94 |
| $^{252}$Cf Maxwellian (ISO) | 2.13 | -3.46 | 1.35 | -6.39 | 1.96 |
| $^{252}$Cf Maxwellian (Grundl) | 2.10 | -2.88 | 1.35 | -6.51 | 1.96 |
| $^{252}$Cf Watt (Froehner) | 2.11 | -3.20 | 1.35 | -8.91 | 4.35 |

**Table 3:** N50 slab neutron efficiency percentage difference between the modelled and experimental results for the $^{252}$Cf and (α,n) sources at 500mm and 1000mm distance. Uncertainties are shown at the 1 standard deviation level.

From Table 3 it can be observed that the difference between the modelled and experimental results at 500mm was within ±5% in most of the cases, except for the Am/B source and the $^{252}$Cf Watt spectrum that produced results within ±8%. As with the $^{252}$Cf presented before, the results were worse when the distance was increased for most of the sources/spectrum combinations studied.

## 7. Angular Dependence

The N50 slab $^{252}$Cf neutron detection efficiency angular dependency has been studied with MCNPX$^{TM}$. This provides valuable, additional characterisation data for the N50 slabs, over and above the characterisation already performed [2, 3]. The data allows assessments to be performed, for the performance of the N50 detectors, for diverse geometries including, for example, special nuclear material holdup monitoring where localised (point-like) sources are expected to occur at a wide range of angular distances from the slabs. We did not collect experimental data in this case for comparison. The spectra used for this study are the Watt and the modified Maxwellian according to Grundl [7]. The sources were modelled at intervals of 5$^{o}$ from the normal to the slab up to 90$^{o}$ for a 500mm and a 1000mm radial distance measured from the mid active region on front face of the slab, sweeping out this way a circular plane at right angles at the detector axis. Figure 1 shows a rather flat response in the central part of the scan with the efficiency only decreasing dramatically for angles larger than about 40$^{O}$ from the normal.

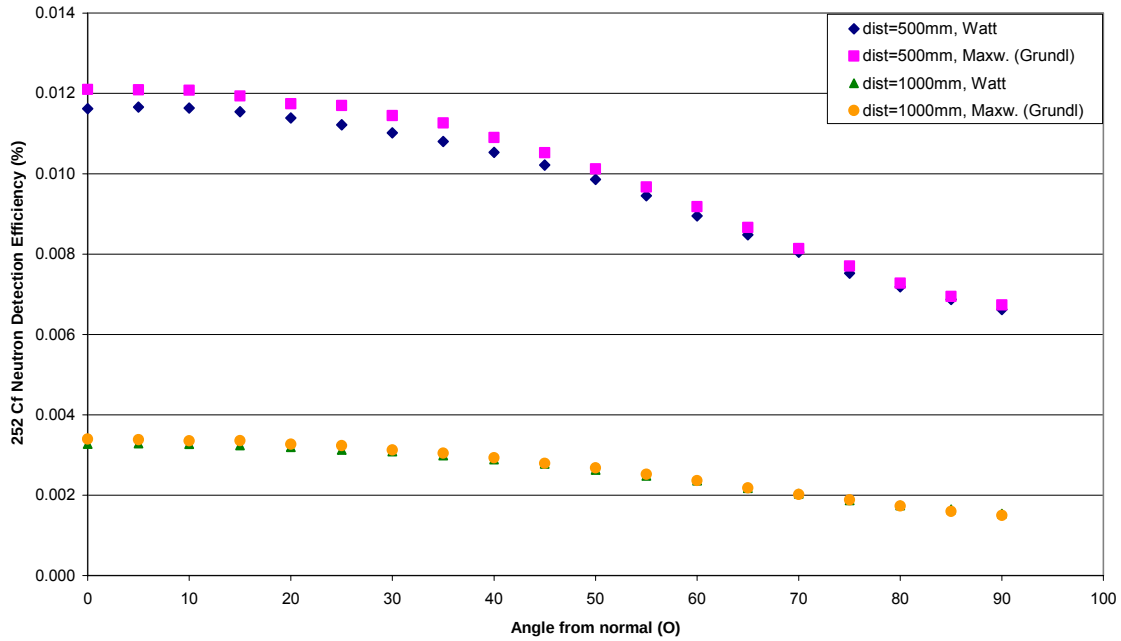


**Figure 4:** N50 slab $^{252}$Cf neutron detection efficiency vs angle from normal.

## 8. Conclusions

The 'simple case' exercise has been extended in this work by including experimental results taken over the range 200mm to 1000mm with the same 'point-like' $^{252}$Cf source. The experiment was modelled using MCNPX$^{TM}$ using different $^{252}$Cf emitted neutron energy spectra. The agreement between the calculated and experimental results was less than 5% for distances smaller than 500mm, where the greatest effort was made to quantify the room return background, although the difference was found to get larger with distance. There was a marked difference in calculated results depending on what spectrum was selected. This underscores the need for care in selecting and benchmarking spectra to the application and also speaks to the need for on-going nuclear data evaluation. It was observed that the results using the Watt spectrum produced consistently worse results across the modelled distances than the other $^{252}$Cf spectra modelled in MCNPX$^{TM}$.

In addition data were gathered at 500mm and 1000mm using certified ($\alpha$,n) sources of Am/Li, Am/F, Am/B and Am/Be. The Monte Carlo study was extended for these sources and various ($\alpha$,n) spectra were assessed. The modelled results agreed with the experimental data at 500mm within ±5% difference in most of the cases, except for the Am/Li/F sources and the $^{252}$Cf Watt spectrum that produced results within ±8% difference. We consider this agreement to be excellent bearing in mind far fewer measurements and evaluations are available for this class of sources compared to $^{252}$Cf.

The N50 slab $^{252}$Cf neutron detection efficiency angular dependency was studied for the first time with MCNPX$^{TM}$ using the Watt representation and the Maxwellian according to Grundl and Eisenhauer's scheme. The sources were at 500mm and a 1000mm radial distance and the response was found to be fairly flat in the central region and only decreasing more than 20% for angles larger than 50$^{O}$ from the normal. This provides valuable, additional characterisation data for the N50 slabs, for diverse measurement applications in safeguards.

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# Enhancement Possibilities to the IMCA Software

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***Abstract:***

*The IMCA is a portable instrument designed to perform uranium enrichment measurements rapidly and with minimum setup using the enrichment meter method. Growing concerns for the possible illicit diversion of nuclear materials has placed a greater emphasis on verification measurements of enriched uranium. If larger inventories of uranium are made more accessible to international inspection the activities of the international and domestic safeguards organizations responsible for establishing and verifying these inventories will increase. In light of this climate there is a need for a review of the capabilities and possible enhancements to this established and extensively used technique. Recent work (by one of the authors) has focussed on improvements to the actual measurement process. The work reported here is focussed on possible enhancements to the software in an effort to minimize the burden on the inspector.*

*IMCA encompasses a software suite of packages tailored toward different detector types; namely IMCG for high resolution germanium detectors, IMCN for sodium iodide detectors, and IMCC for cadmium zinc telluride detectors. Initial reviews and enhancements are evaluated here for the IMCG package, but an extension to the other packages is straightforward. Included in the review is a consideration of expanding the choice of available matrix materials and container wall materials. Specifically Ni has been added as an optional wall material. Another aspect reviewed is the measurement uncertainty and the relative importance of the various contributing factors. For example the software assumes a fixed measurement uncertainty on the wall thickness, but perhaps allowance should be made for this value to be adjustable based on the calibration of the ultrasonic probe and the scatter in the observations. At present no allowance is made for possible variations in the chemical compositions of the special nuclear materials being measured, or in the inherent uncertainties in supposedly 'known' physical constants. Though expected to be small, these variations could be captured in a reported Total Measurement Uncertainty (TMU). The presence of interfering surface layers in typical $UF_6$ cylinders may be accounted for by the option of introducing in software an additional attenuator either directly or as part of the TMU. Finally, a 'count to precision' option would automatically enable an early stop or an extended count as appropriate, when a particular precision criterion must be met, rather than requiring the inspector expend excessive time or conversely perform a repeat measurement.*

**Keywords:** uranium, enrichment, IMCA, IMCG, $UF_6$.

## 1. Introduction

The IMCA software is used to determine the fraction of fissile $^{235}U$ to total uranium using the enrichment meter technique ([1] and references therein). The software [2] was developed according to IAEA guidelines and follows the IAEA nomenclature for similar packages. The IMCG software is the version intended for use with germanium (Ge) detectors where typically a 15 mm thick, 500 $mm^2$ front surface area planar crystal most suited to low energy gamma rays is employed. The enrichment level of $^{235}U$ is ascertained by counting the 186 keV gamma rays emitted (the highest intensity emission). The technique is intended for use with fresh bulk material held in containers with wall thicknesses up

to about 15mm of steel. The typical applications are international safeguards inspections for verifying uranium stock, and in process/product control in enrichment and fuel fabrication plants.

In recent work [3, 4] several issues typically encountered in measurement situations have been discussed and evaluated. By taking suitable care in the control of the experimental conditions (detector placement, wall thickness measurement, avoidance of surface anomalies, and avoidance of $UF_6$ void area) the IMCG portable gamma spectrometry system [5] may be applied to the enrichment measurement of storage containers to give uncertainties (both precision and bias) which are significantly lower than are either typically reported in the literature or embodied in the safeguards target value guides. The large cylinders studied [3] were of types 30B, 48X, 48Y, and 48G (wall thicknesses ranging from 2/8" to 5/8") and contained $DUF_6$, $NUF_6$ and $LEUF_6$. This work is noteworthy in that it summarizes the results from 1767 assays which is a much larger number of assays than usually reported in such studies, and it is also based on actual containers rather than laboratory test cases.

This present work discusses the source of the measurement uncertainties viewed from a calculational perspective.

## 2. Enrichment Calculation

In order to evaluate the uncertainty contributions due to the various factors described above it is necessary to review the way in which the enrichment is calculated. This section presents the underlying algorithms for the calculation of the enrichment and uncertainty. The next section will discuss the impact of the various contributors to the overall uncertainty.

The enrichment determination in IMCG is made using the following equation:

$$E_a = K_{bare} \cdot R \cdot F_{material} \cdot F_{container} \qquad (1)$$

Here $E_a$ is the enrichment value in atom percent, $K_{bare}$ is the average calibration constant determined over a set of reference items and corrected to the bare (non-encapsulated) material, $R$ is the count rate for the '186 keV' peak, $F_{material}$ is a correction factor for the bulk material composition, and $F_{container}$ is a correction factor for the item container wall. Note that as a convenience we refer to the 186 keV peak although as will be made clear below a region of interest encompassing both the weak 183 keV line and the much stronger 186 keV line from $^{235}U$ is used.

The corresponding uncertainty in the enrichment at the standard deviation level is evaluated according to:

$$\sigma_{E_a} = E_a \sqrt{\left(\frac{\sigma_{K_{bare}}}{K_{bare}}\right)^2 + \left(\frac{\sigma_R}{R}\right)^2 + \left(\frac{\sigma_{F_{container}}}{F_{container}}\right)^2} \qquad (2)$$

where the subscripts on the individual $\sigma$ terms indicate the origin of the uncertainty. Note that there is currently no uncertainty contribution carried through from the factor $F_{material}$.

The enrichment meter principle relies on the measured items having 'infinite thickness'. This concept is briefly reviewed here in order to put the discussion in context since there are other techniques for determining uranium enrichment [6], albeit with different measurement condition requirements.

For most practical purposes 'infinite thickness' may be defined as the thickness for which the error in assuming an infinite sample size is < 0.1%; for high accuracy work uncertainties at this level might call for attention. For a uniform slab geometry viewed in the far-field approximation the self attenuation factor is given by *[1-exp(x)]/x* where *x* is the 'optical thickness' in units of mean free paths (mfp's). The term *1-exp(x)* can also be thought of as the sample volume contribution to the total count rate. From

this we note that ~7 mfp's are needed to satisfy the thick specimen assumption. Table 1 lists the "infinite thickness" for some typical uranium compounds. Values for the linear attenuation coefficient μ were obtained from [7]. Table 2 lists the values of *1-exp(x)* for various sample depths of different uranium compounds. Figure 1 illustrates what this means in practice. For the typical compounds listed a sample depth of 30 mm (3cm) satisfies the infinite thickness criterion.

This in turn means that beyond this thickness the sample is not 'seen' and so necessarily an assumption of homogeneity of the sample must be assumed, unless already known from process knowledge of a destructive assay technique. The consequence for the measurement is that it is of greater importance to ascertain the correct uncertainties, particularly if repeat measurements are not practicable.

| "Infinite Thickness" for common uranium compounds | | | | |
|---|---|---|---|---|
| Uranium Compound | Density ($gcm^{-3}$) | μ at 186 keV ($cm^{-1}$) | Mean Free Path (1/μ) (cm) | "Infinite Thickness" 7 mfp's (mm) |
| U Metal | 18.70 | 27.30 | 0.037 | 2.56 |
| $UO_2$ (sintered) | 10.90 | 14.17 | 0.071 | 4.94 |
| $U_3O_8$ (powder) | 7.30 | 9.13 | 0.110 | 7.67 |
| $UF_6$ (solid) | 4.70 | 4.79 | 0.209 | 14.60 |
| Uranyl Nitrate | 2.80 | 2.16 | 0.463 | 32.38 |
| $UO_2$ (powder) | 2.00 | 2.60 | 0.385 | 26.92 |

**Table 1:** "Infinite Thickness" for common uranium compounds.

| Infinite thickness & Sample depth | | | | | |
|---|---|---|---|---|---|
| Uranium Compound | Volume contribution to count rate for various sample depths D (mm) [1-exp(-μD)] | | | | |
| | 1 | 5 | 10 | 15 | 30 |
| U Metal | 0.9348 | 1.0000 | 1.0000 | 1.0000 | 1.0000 |
| $UO_2$ (sintered) | 0.7576 | 0.9992 | 1.0000 | 1.0000 | 1.0000 |
| $U_3O_8$ (powder) | 0.5985 | 0.9896 | 0.9999 | 1.0000 | 1.0000 |
| $UF_6$ (solid) | 0.3808 | 0.9090 | 0.9917 | 0.9992 | 1.0000 |
| Uranyl Nitrate | 0.1944 | 0.6607 | 0.8849 | 0.9609 | 0.9985 |
| $UO_2$ (powder) | 0.2289 | 0.7275 | 0.9257 | 0.9798 | 0.9996 |

**Table 2:** "Infinite Thickness" and sample depth for common uranium compounds.

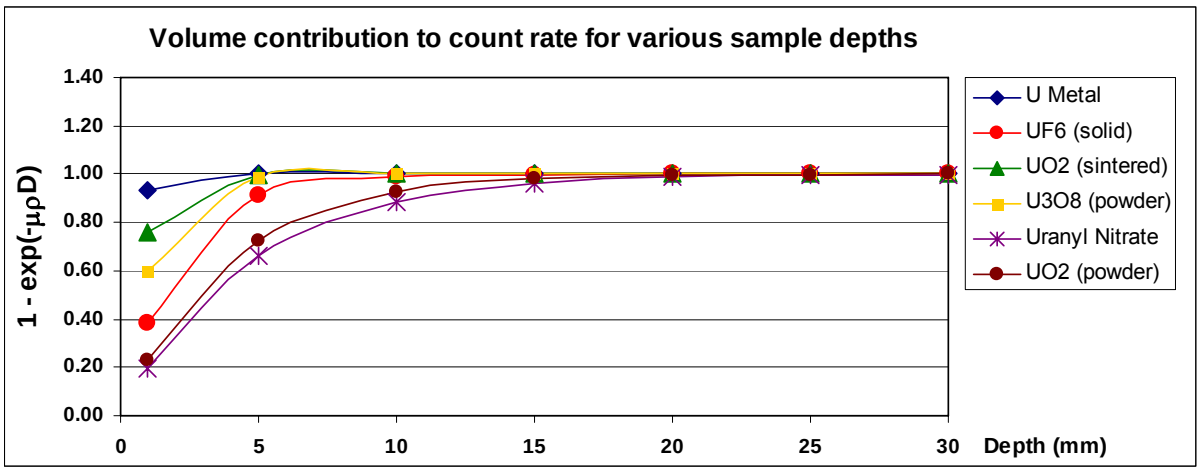


**Figure 1.** Volume contribution to count rate for various sample depths D (mm).

## 2.1. Count Rate

The count rate in the peak intensity is determined using a region of interest (ROI) method with one ROI defined around the peak itself, and two continuum ROIs on either side of the peak used to obtain an estimate of the 'background' under the peak. Once the 186 keV peak has been identified the ROIs

are automatically defined by the software during setup using the following rules based on the peak position (prescription for the energy calibration) and actual full-width-half-max (FWHM) of the peak:

- Beginning channel for peak ROI: 182.6 keV – 2 x (FWHM of 186 keV peak
- End channel for peak ROI: 185.7 keV + 2 x (FWHM of 186 keV peak)

If the channels defining the range of the peak ROI are referred to as “left” and “right”, the background ROIs are defined as:

- Low background ROI (in channels): (“left” – 13) to (“left” – 3)
- High background ROI (in channels): (“right” + 3) to (“right” + 13)

Figure 2 illustrates the ROI definitions for a sample spectrum. The region highlighted in red shows the peak ROI and the regions in blue indicate the background regions. These are chosen to be outside the influence of the lines being used for quantification in regions free from potential interference.

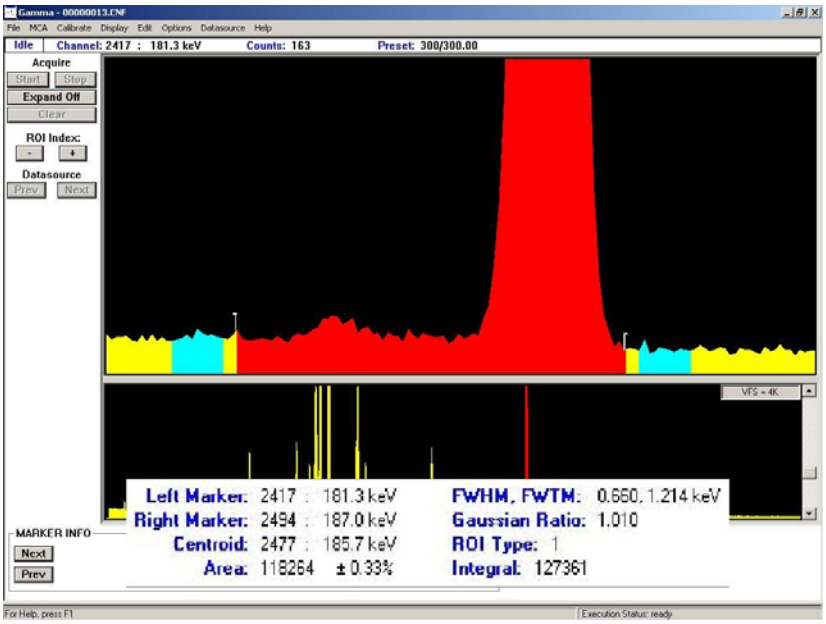


**Figure 2:** Example spectrum showing ROIs used for evaluation of 186 keV gamma count rate.

The net count rate for the 186 keV peak is given by:

$$R = \frac{G - \frac{N}{N_B}(B_1 + B_2)}{t} \qquad (3)$$

where $G$ is the sum of counts in the peak ROI (including the end point channels), $N$ is number of channels in the peak ROI, $N_B$ is the number of channels in the low and high background ROIs (i.e. the sum of channels in both ROIs which is equal to 22), $B_1$ is the sum of counts in the low background ROI (including the end point channels), $B_2$ is the sum of counts in the high background ROI (including the end point channels), and $t$ is the acquisition live time in seconds.

The count rate uncertainty (one standard deviation precision) is given by:

$$\sigma_R = \frac{\sqrt{G + (N / N_B)^2 (B_1 + B_2)}}{t} \qquad (4)$$

## 2.2 Container Correction Factor

In order to account for the attenuating effects of the container a container wall correction factor $F_{container}$ is applied to the measured rate. $F_{container}$ is given by:

$$F_{container} = e^{\mu_{SampleContainer} t_{SampleContainer}} \quad (5)$$

where $\mu_{SampleContainer}$ is the linear gamma attenuation coefficient of the sample container wall at 186 keV in units of $cm^{-1}$, and $t_{SampleContainer}$ is the thickness of the sample container wall in cm.

The linear gamma attenuation coefficients that are supported in the software are given in Table 1. These values were part of the original specification of the instrument. A convenient reference source of evaluated gamma interaction data is the NIST library [7]. Combined with material properties the entries in Table 3 can be compared with XCOM values.

| Container Material Correction Factors | |
|---|---|
| Material Composition | Linear Gamma Attenuation Coefficient ($cm^{-1}$) |
| Steel | 1.210 |
| Aluminum | 0.340 |
| Zircalloy-2 | 1.640 |
| Polyethylene | 0.132 |
| Monel | 1.480 |
| Glass | 0.312 |
| Nickel | 1.390 |

**Table 3:** Container Material Correction Factors.

The uncertainty in the container correction factor is evaluated in the software by:

$$\sigma_{F_{container}} = F_{container} \sqrt{(t_{SampleContainer}\sigma_{\mu})^2 + (\mu_{SampleContainer}\sigma_t)^2} \quad (6)$$

where $\sigma_t$ is the uncertainty in the container wall thickness, and $\sigma_\mu$ is the uncertainty in the attenuation coefficient. These uncertainties are currently held fixed with $\sigma_\mu$ set equal to 0.01 of $cm^{-1}$ and $\sigma_t$ set equal to 0.1 mm [8]. A 'better' practice might be to take a fixed proportionate uncertainty in the linear attenuation coefficients or to use material specific values. For the wall thickness uncertainty a measured value specific to the item being measured would seem more appropriate.

## 2.3 Material Correction Factor

The material correction factor $F_{material}$ is used to account for the possible difference between the matrix of the calibration standard and the measured item. For example, a typical calibration standard material is $U_3O_8$, whereas a typical measured item is $UF_6$.

$F_{material}$ is defined in terms of a matrix factor defined for each uranium compound:

$$F_{material} = \frac{F}{F_S} \quad (7)$$

where $F$ and $F_S$ are the matrix factors for the measured item and the calibration standard respectively. The matrix factor for each compound is defined as a ratio in reference to uranium metal which in the far field approximation for an infinite thickness of homogeneous material can be written as:

$$F = \frac{\mu_\rho}{\mu_{\rho U} W_U} \tag{8}$$

where $\mu_\rho$ is the mass attenuation coefficient (MAC) for the compound at 186 keV, $\mu_{\rho U}$ is the MAC for uranium at 186 keV, and $W_U$ is the weight fraction of uranium (≤1) in the item. (The MAC is the ratio of the linear attenuation coefficient and material density.)

For example, if a calibration is performed using $U_3O_8$ standards, and the user wishes to measure $UF_6$ samples, $F_S$ would be 1.0096 ($\mu_\rho$=1.250 g.cm$^{-2}$; $\mu_{\rho U}$=1.460 g.cm$^{-2}$; $W_U$=84.80%) and $F$ would be 1.0332 ($\mu_\rho$=1.020 gcm$^{-2}$; $\mu_{\rho U}$=1.460 gcm$^{-2}$; $W_U$=67.62%). A correction factor $F_{material}$ of 1.023 would then be applied to the enrichment value obtained. (In this illustration values for the linear gamma attenuation coefficient were taken from [7], and the density values were taken from [1]).

Values for $F_{material}$ for several item/standard combinations are stored *a priori* in the IMCA software and are applied directly based on user choice of compound. The compounds accommodated are U, UC, UN, $UC_2$, $UO_2$, $U_3O_8$, $UF_4$, $UF_6$, and $UO_2(NO_3)_2$, and the values for $F_{material}$ are obtained from the specification in [8] where a table of values is given for different combinations.

An uncertainty contribution arising from $F_{material}$ is not currently propagated in the final enrichment determination. This is a potential source of systematic bias that at present is neglected.

## 2.4 Calibration Constant

In order to tie in the detector efficiency and measurement geometry with the measured count rate a calibration measurement must be performed. While in principle a single calibration measurement will suffice it is good practice to perform several calibration measurements using standards of varying enrichment that typically span the expected range of enrichment in the measurements to be performed.

The calibration constant $K$ is calculated as:

$$K = \frac{E_a}{R \cdot F_{container}} \tag{3}$$

where the enrichment of the standard is known as is the container material and thickness for the standard container. The uncertainty in $K$ is given by:

$$\sigma_K = K\sqrt{\left(\frac{\sigma_{E_a}}{E_a}\right)^2 + \left(\frac{\sigma_R}{R}\right)^2 + \left(\frac{\sigma_{F_{container}}}{F_{container}}\right)^2} \tag{4}$$

If multiple standards are available, each measured standard yields a value for $K$, and an average calibration constant $K_{average}$ is then obtained from the weighted average of the individual values.

Once the calibration constant has been obtained through measurement of the standards a *bare* constant, $K_{bare}$, is evaluated so that the attenuating effects of the container walls of the calibration standard are removed. $K_{bare}$ is given by:

$$K_{bare} = Ke^{-\mu_{SampleContainer} t_{SampleContainer}} \tag{5}$$

and the uncertainty is given by

$$\sigma_{K_{bare}} = \sigma_K e^{-\mu_{SampleContainer} t_{SampleContainer}} \tag{6}$$

In cases where the sample is known to have thicker container walls than the calibration standards, an intervening wall material of similar composition is introduced during the calibration measurements in order to mimic the sample measurement geometry.

## 3. Discussion of Uncertainties

### 3.1. Count Rate

The choice of ROIs was part of the original specification of the IMCA where presumably simplicity and robustness were the key criteria. An ROI analysis is straightforward especially when it comes to evaluating the statistical precision since in essence it is a basic counting experiment. A broad peak region was selected which makes the analysis insensitive to possible slight gain drifts, and by including the 183 keV peak the resulting area is insensitive to resolution degradation that may occur in an electrically of mechanically noisy environment. The continuum ROIs have also been placed in featureless regions.

A wide ROI, however, also includes more continuum than strictly necessary and so the precision on the net peak area will be impaired. Optimizing the precision for a given acquisition is important for minimizing the overall uncertainty and it is also important if an effective count-to-precision algorithm is to be implemented. In addition to or instead of the ROI analysis, a doublet peak fitting algorithm is proposed. Standard spectroscopy tools can perform this analysis, making better use of the known energy difference, peak shape, and relative intensity of the two peaks in the doublet. In principle if a library directed peak fitting routine were used the 186 keV line alone could be used with the benefit of an improved Signal-to-Noise ratio by virtue of encompassing fewer counts in the narrower energy band that would be needed to characterize the peak.

An additional consideration in the current application is the impact of including the 183 keV line in the ROI on the selection and use of the MAC for the 186 keV line. This was evaluated based on the relative intensities of the two lines. The 185.715(5) keV line from $^{235}U$ has an abundance (or branching ratio) of 57.2(5)%, while the 182.61(5) keV line has an abundance of 0.34(2)% [9]. (Here the standard deviation is quoted in the least significant figure in brackets.) An intensity weighted average [$=(I_1.U_1 + I_2.U_2) / (I_1 + I_2)$] (where the nomenclature is self explanatory) can be used to evaluate the impact. Taking $UF_6$ as an example we have:

- MAC at 182.6 keV: 1.07 $cm^2.g^{-1}$
- MAC at 185.7 keV: 1.03 $cm^2.g^{-1}$
- Weighted Mean: = (0.34*1.07 + 57.2*1.03) / (0.34 + 57.2) = 1.0302 $cm^2.g^{-1}$

It is seen that because of the closeness of the energies and hence the similarity in the MACs, forming the intensity weighted mean value returns, for all practical purposes, the MAC for the 186 keV line. The impact of the 183 keV line on the selection of MAC data is therefore negligible by comparison with the impact from other factors.

### 3.2. Container Correction Factor

The current specification for the uncertainty in $\sigma_\mu$ is 0.01 $cm^{-1}$. Better practice for $\sigma_\mu$ might be to take a fixed proportionate uncertainty in the linear attenuation coefficients or to use material specific values. If a 1% uncertainty in $\sigma_\mu$ is to be considered reasonable (discussion in next section), the 0.01 $cm^{-1}$ choice is seen to be an appropriate upper estimate when compared with the values of $\mu$ given in table 3.

In the present implementation the uncertainty in the container thickness ($\sigma_t$) is assumed to be the same for all wall materials and wall thicknesses. In practice the wall thickness is measured for each item using an ultra-sonic gauge so that the particular gauge used and the measurement position are two sources of uncertainty that could be used by the software to estimate a refined uncertainty estimate. While the uncertainty can be better estimated using sampling measurements at various locations on the sample [3], measurement time is often a limitation particularly during safeguards inspections. A refined uncertainty estimate from the software can then be an advantage. Checks of

gauges imply a thickness determination of circa 0.08mm (+/-0.1mm in 3mm is stated as leading to a 2% relative error on the enrichment determination at 5wt%) and under laboratory conditions a 0.03 mm uncertainty can be attained. A 0.1 mm uncertainty is therefore a reasonable choice.

If the container wall (material, thickness and curvature) is mimicked during calibration any uncertainty in the correction factor error is effectively reduced because allowance is only needed for the *difference* between the two measurement conditions. Table 4 shows the percentage increase in the uncertainty in $F_{container}$ when $\sigma_t$ is set equal to 0.2 mm instead of 0.1 mm (the current specification).

| $F_{container}$ & ***difference*** in $\sigma_{Fcont.}$ (i.e. $\Delta\sigma_{Fcont}$ in %) as a function of material and wall thickness. ($\sigma_{Fcont.}$ going from 0.1 mm to 0.2 mm.) | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|
| Material | $\mu$ ($cm^{-1}$) | Container wall thickness (mm) | | | | | | | | | |
| | | 1 | | 5 | | 10 | | 15 | | 20 | |
| | | $F_{cont.}$ | $\Delta\sigma_{Fcont.}$ | $F_{cont.}$ | $\Delta\sigma_{Fcont.}$ | $F_{cont.}$ | $\Delta\sigma_{Fcont.}$ | $F_{cont.}$ | $\Delta\sigma_{Fcont.}$ | $F_{cont.}$ | $\Delta\sigma_{Fcont.}$ |
| Polyethylene | 0.132 | 1.01 | 0.03 | 1.07 | 0.01 | 1.14 | 0.00 | 1.22 | 0.00 | 1.30 | 0.00 |
| Glass | 0.312 | 1.03 | 0.13 | 1.17 | 0.03 | 1.37 | 0.01 | 1.60 | 0.01 | 1.87 | 0.01 |
| Aluminum | 0.340 | 1.03 | 0.15 | 1.19 | 0.03 | 1.40 | 0.02 | 1.67 | 0.01 | 1.97 | 0.01 |
| Steel | 1.210 | 1.13 | 1.05 | 1.83 | 0.41 | 3.35 | 0.22 | 6.14 | 0.15 | 11.25 | 0.11 |
| Nickel | 1.390 | 1.15 | 1.24 | 2.00 | 0.53 | 4.01 | 0.28 | 8.04 | 0.19 | 16.12 | 0.14 |
| Monel | 1.480 | 1.16 | 1.34 | 2.10 | 0.60 | 4.39 | 0.32 | 9.21 | 0.22 | 19.30 | 0.16 |
| Zircalloy-2 | 1.640 | 1.18 | 1.51 | 2.27 | 0.72 | 5.16 | 0.39 | 11.70 | 0.27 | 26.58 | 0.20 |

**Table 4:** Container wall correction factor & ***difference*** in uncertainty $\sigma_{Fcont.}$ ($\Delta\sigma_{Fcont}$ in %) as a function of material and wall thickness ($\sigma_{Fcont.}$ going from 0.1 mm to 0.2 mm).

As expected the relative impact is greatest at the smallest wall thicknesses for the materials with the largest values of $\mu$. The effect is a 1% increase in the uncertainty for an increase in wall thickness uncertainty of 0.1 mm.

### 3.3 Material Correction Factor

As noted previously there is no uncertainty associated with $F_{material}$ in the current specifications. Estimating the absolute accuracy of MACs is difficult and the authors are unaware of a comprehensive treatment covering the energy range and materials of interest here. By drawing on a selection of evaluations published over the years based on judging the scatter resulting from different treatments of the various experimental determinations and associated supplemental theoretical treatments (and sometimes ad hoc adjustments), a reasonable (but perhaps subjective) estimate may be made on the uncertainty in MAC values.

Based on the available literature and review by [10] and subject to the interpretation of the authors, it is felt an RSD of ~1% may be assumed for the medium atomic number metals (structural materials), while for the uranium compounds a RSD of ~2% seems more appropriate. (From material to material, however, there will be strong correlations--in other words the majority of the uncertainty is systematic rather than random for materials close in atomic number.)

An estimate on the uncertainty in $F_{material}$ based on the values above yields an effect of the order of ~0.1% for typical calibration/sample materials, but should be evaluated on a case by case basis. An additional lookup table could be added to the software for the estimated uncertainties associated with $F_{material}$.

The addition of other sample materials such as $UF_4$ is also considered as an enhancement.

### 3.4 Calibration Constant

It is most important to use multiple, well-characterized, calibration standards when possible in order to obtain a suitable average, to span the range of enrichments to be counted, and to mimic the expected measurement geometry as closely as possible during the characterization measurements. Traditionally the availability of standards has not been an issue, but it is conceivable that in the future

this may become a limitation. The measurement geometry can be more of an issue as discussed in [3].

In principle if a characterized detector is used, a calculational tool such as ISOCS [11] could be used to obtain K, but given the uncertainty in basic physical data such as the specific photon yield this would be less accurate than using standards. If representative standards are not available, however, a tool such as ISOCS may be useful in transferring the calibration from standards that do exist to another unusual geometry and also to generate a realistic and justifiable uncertainty estimate.

## 3. Summary

The IMCA method of non destructively determining the uranium enrichment of bulk materials in field measurements has been reviewed. Previous work has shown how careful selection and control of the measurement conditions can directly benefit performance. In this work we have highlighted a number of possible refinements to the data acquisition and analysis software to ease the burden on the user and also yield results with a more comprehensive uncertainty treatment.

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# Predicting the performance limit of relative isotopic codes for the determination $^{240}Pu_{eff}$

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***Abstract:***

*The quantification of plutonium via passive correlated neutron counting requires knowledge of the relative isotopic composition because several Pu isotopes undergo spontaneous fission. Traditionally the strength of the spontaneous fission signal is expressed in terms of the effective mass of $^{240}Pu$ present. This is because $^{240}Pu$ is typically the dominant contributor for materials in the civilian reactor and weapons fuel cycle. The effective $^{240}Pu$ mass is a weighted linear sum of the fertile isotopes. A non destructive determination of the $^{240}Pu_{eff}$ weight fraction may be made by analysing the gamma ray spectrum of the item acquired using a high resolution spectrometer. In particular the 100keV region is key since it contains the 104 keV line from $^{240}Pu$, which is often the sole viable means of quantifying its relative abundance, mingled in with other spectral features.*

*The question we address in this work is how to predict whether a relative isotopic determination of a particular item with a particular set-up is likely to be viable. This is akin to asking what the minimum detection limit is. It is important to know this when designing and specifying new waste assay systems, for example, or when planning measurement sequences.*

*The pragmatic approach we have taken is to analyse a wide variety of spectra collected using a set of PIDIE sources under different conditions of attenuation and collection time. The spectra were analysed and a plot of the Relative Standard Deviation (RSD) reported for the $^{240}Pu_{eff}$ weight fraction determination was constructed as a function of the net number of counts in the 100 keV complex. We consider the detection limit to correspond to a RSD of about 30% in line with the familiar concepts of Lloyd Curie. By this approach we use actual experimental data to obtain an estimate of the number of counts needed in the 100 keV region for success. For an unknown case we predict the count rate that may be observed by using the ISOCS efficiency estimating tool seeded with the decay data for the various Pu isotopes. If the predicted rate for the measurement configuration considered, which may be a waste container, say, results in a count that exceeds the number required in the data acquisition period allowed we can have confidence that the isotopic code will run and yield reliable results.*

**Keywords:**MGA, relative isotopics code, gamma spectroscopy

## 1. Introduction

The quantification of plutonium is an important aspect of non-destructive assays of special nuclear materials. Several techniques are used. One of them being passive neutron coincidence counting. This technique exploits the fact that multiple neutrons from spontaneous fission are emitted essentially simultaneously. In general, plutonium usually contains several isotopes that have large spontaneous fission yields: $^{238}Pu$, $^{240}Pu$, and $^{242}Pu$, so the observed coincidence response is due to all three isotopes. $^{240}Pu$ is typically the dominant contributor for materials in the civilian reactor and weapons

fuel cycle. So, traditionally the strength of the spontaneous fission signal is expressed in terms of the effective mass of $^{240}$Pu present, which can be calculated as a weighted linear sum of all three isotopes. $^{240}Pu_{eff}$ weight fractions are used in the interpretation of passive neutron coincidence counter results. A non destructive determination of the $^{240}Pu_{eff}$ weight fraction can be done by analysing the gamma ray spectrum of the item acquired using a high resolution spectrometer. There are only three gamma lines that carry $^{240}$Pu information (104, 160, 642 keV) and often the high energy lines can be at background levels. So, the 100 keV region is key since it contains the 104 keV line from $^{240}$Pu, which is often the sole viable means of quantifying its relative abundance. This is a challenging objective given the complexity of the spectrum from plutonium materials in this energy region.

Analysis of plutonium spectra is usually performed using the isotopic codes. One of them is the MGA code which was developed by Ray Gunnink and co-workers at the Lawrence Livermore National Laboratory [1]. This code analyses a 100 keV region containing a number of strongly overlapping peaks. The precision of an analysis is strongly dependent on quality of the spectrum, which is, for the detector with good energy resolution, strongly influenced by the counting statistics in the 100 keV region. In this work we have taken the pragmatic approach by analysing a wide variety of spectra collected using a set of Plutonium Isotopic Determination Intercomparison Exercise (PIDIE) sources [2] under different conditions of attenuation and collection time. The RSD for each measurement was then evaluated as a function of counting statistics in the 100 keV region.

## 2. Experimental Set-up

Measurements were taken using a set of sources (PIDIES), which vary in isotopic content, from low to high burn-up, using a BEGe detector (Canberra Model BE3820) and a digital spectrum analyzer (Canberra Model DSA2000). The isotopic composition of the samples is given in Table 1 [3]. The BEGe detector was set to use a gain of 0.095 keV/ch for a 16184 channel spectrum, which is a generally used value for the radioactive waste assay systems to cover a list of general radionuclides in the energy range from 60 to 1500 keV. The resolution (FWHM) for the measurements was approximately 710 eV at 122 keV. The PIDIE sources were hold by a retort stand with the gamma-ray window facing the detector at 42cm distance, which is similar to the detector to the centre of a waste drum distance for a typical waste assay system. Filters were placed in front of the detector in order to reduce the intensity of the 59.5 keV line from $^{241}$Am: tin filter with 1.14 g·cm$^{-2}$ areal density and a copper plate with 1.05 g·cm$^{-2}$ areal density to form a graded shield.

| | **Weight percent of Pu and Am-241 with respect to $^{tot}$Pu as of April 30, 2006** | | | | | | **Mass $^{tot}$Pu, g** |
|---|---|---|---|---|---|---|---|
| | **Pu-238** | **Pu-239** | **Pu-240** | **Pu-241** | **Pu-242** | **Am-241** | |
| **PIDIE1** | 0.009 | 93.927 | 5.986 | 0.043 | 0.035 | 0.371 | 0.380 (6) |
| **PIDIE2** | 0.018 | 89.677 | 10.113 | 0.097 | 0.095 | 0.593 | 0.411 (4) |
| **PIDIE3** | 0.037 | 85.282 | 14.227 | 0.218 | 0.236 | 1.365 | 0.420 (6) |
| **PIDIE4** | 0.086 | 78.911 | 20.022 | 0.405 | 0.576 | 2.941 | 0.387 (5) |
| **PIDIE5** | 0.104 | 77.196 | 21.531 | 0.455 | 0.714 | 3.287 | 0.378 (7) |
| **PIDIE6** | 0.758 | 69.394 | 24.927 | 1.198 | 3.723 | 7.845 | 0.380 (5) |
| **PIDIE7** | 1.032 | 65.536 | 26.995 | 1.488 | 4.948 | 8.764 | 0.383 (5) |

**Table 1** Isotopic composition of the plutonium samples. The values in brackets are standard deviations on the last significant figures.

Spectra were taken for each of the 7 PIDIE samples alone and with different absorbers. The absorbers used for the measurements were high density (0.96 g·cm$^{-3}$) polyethylene disks with total thickness of up to about 100 mm, and a stack of 10 aluminium plates each of 1.3 mm thick. The duration of the measurements varied so that the number of counts in the 129 keV peak from $^{239}$Pu ranged from approximately 200 to 5000 counts. Further measurements were repeated under the same counting conditions, but in the presence of a $^{137}$Cs source placed directly on top of the detector to simulate high background from fission products. Example of a typical spectrum with and without $^{137}$Cs background is shown on Fig. 1.

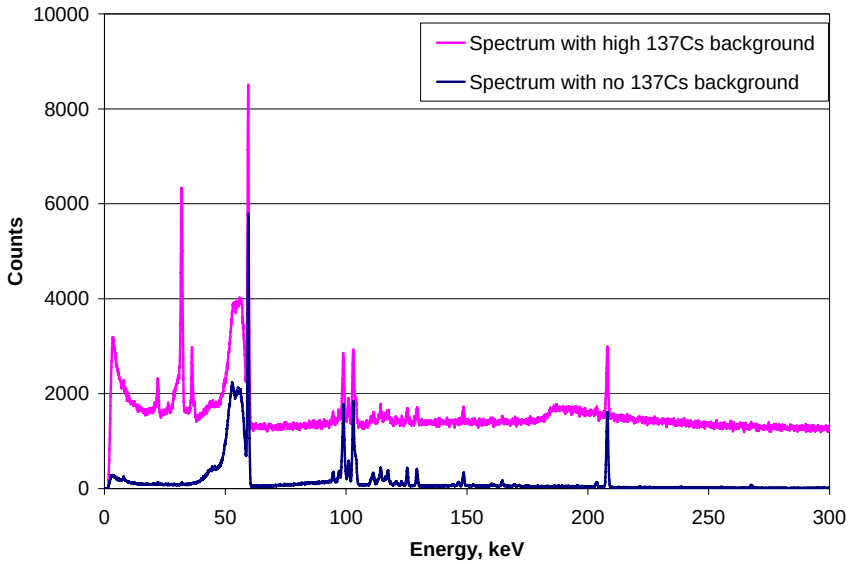


**Fig. 1** 'Typical' plutonium spectra taken with and without $^{137}$Cs background.

## 3. Results

A total of 183 spectra were collected under different conditions of attenuation, background and collection time. All spectra then were analysed with two versions of one of the widely used isotopic codes: the most common and commercially available MGA ver.9.63H and the most recent and improved MGA ver.10. The spectra were analysed and a plot of the Relative Standard Deviation (RSD) reported for the $^{240}Pu_{eff}$ weight fraction determination was constructed as a function of the net number of counts in the 100 keV region. $^{240eff}$Pu weight fractions are used in the interpretation of passive neutron coincidence counter results and therefore it's very important to determine this value with good precision and accuracy. We consider the detection limit to correspond to a RSD of about 30% in line with familiar concepts of Lloyd Curie [4].

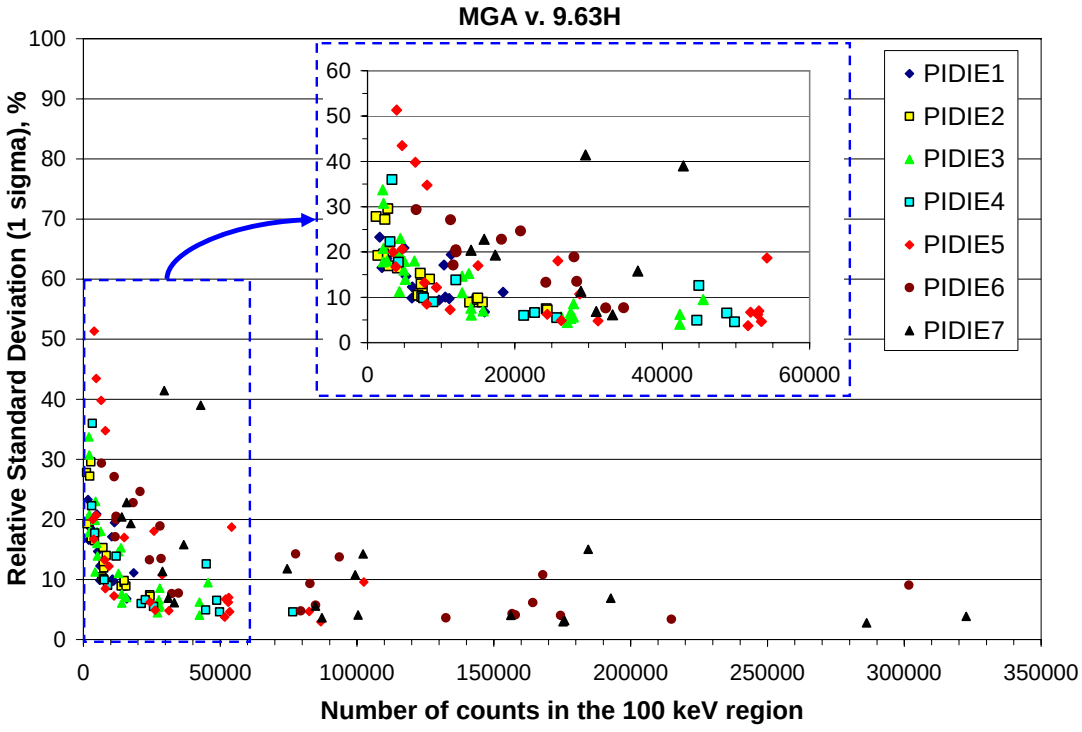


**Fig. 2** RSD for the $^{240eff}$Pu weight fraction determined with MGA ver.9.63H

In a pure counting experiment the minimum detectable activity or lower limit of detection may be defined as the amount of material required so that the probability of a false negative is equal to the probability of a false positive and both are equal to 5%. At this level the RSD on the net signal is ~ 30% and we take this as a reasonable Figure of Merit for the present assessment. By this approach we use actual experimental data to obtain an estimate of the number of counts needed in the 100 keV region for success. Fig. 2 (above) and Fig. 3 represent the results obtained for both versions of MGA. The RSD is that reported by MGA

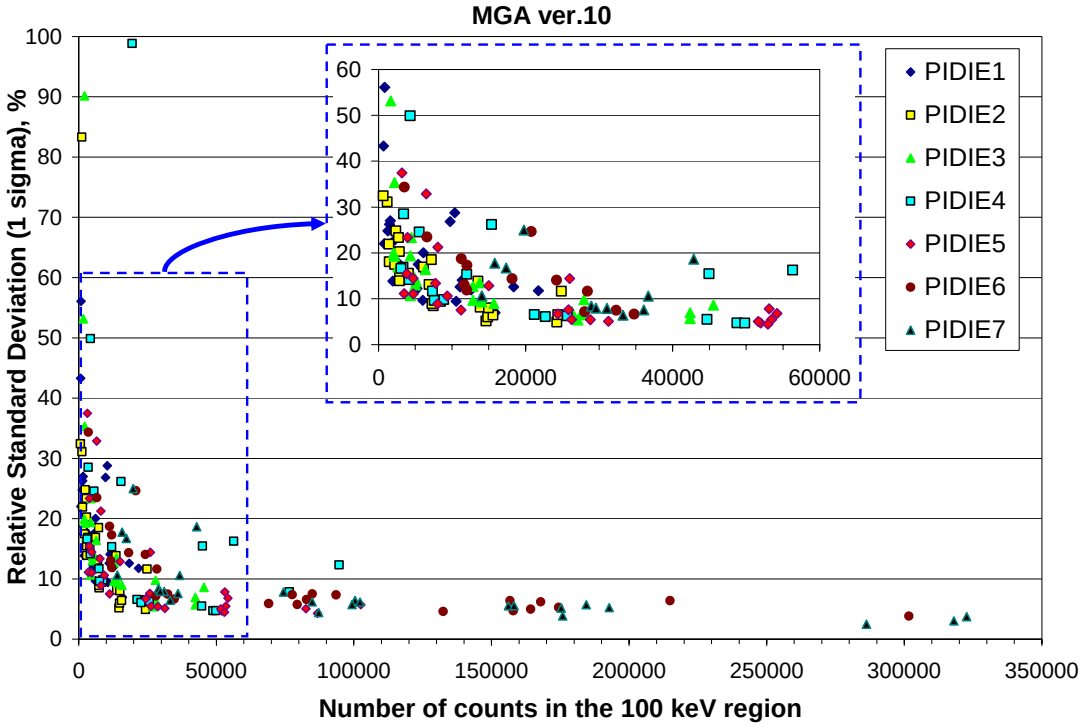


**Fig. 3** RSD for the $^{240eff}$Pu weight fraction determined with MGA ver.10

The results showed that, generally, in order to achieve a 30% counting precision for $^{240}Pu_{eff}$ mass, it was necessary to have no more than about 7000–8000 counts in the 100 keV region. Although the minimum value in the 100 keV region is strongly dependent on counting conditions (i.e. gamma attenuation, presence of a background, isotopic composition) and may significantly vary from spectrum to spectrum. For example, for low burn-up samples, with no high background present, 3500-4000 counts was enough to achieve a desired counting precision. Whereas for high burn-up samples the 104 keV peak from $^{240}$Pu was strongly interfering with $^{241}$Am gamma lines and Pu X-rays, and that required more counts in the 7000-8000 range for acceptable analysis. It should also be noted that MGA ver.9.63H was developed for use primarily in safeguard community. It has strict internal requirements to the quality of the analysed spectrum. As a result, for 34 out of the 183 spectra, MGA ver.9.63H did not produce any results, see Fig. 4.

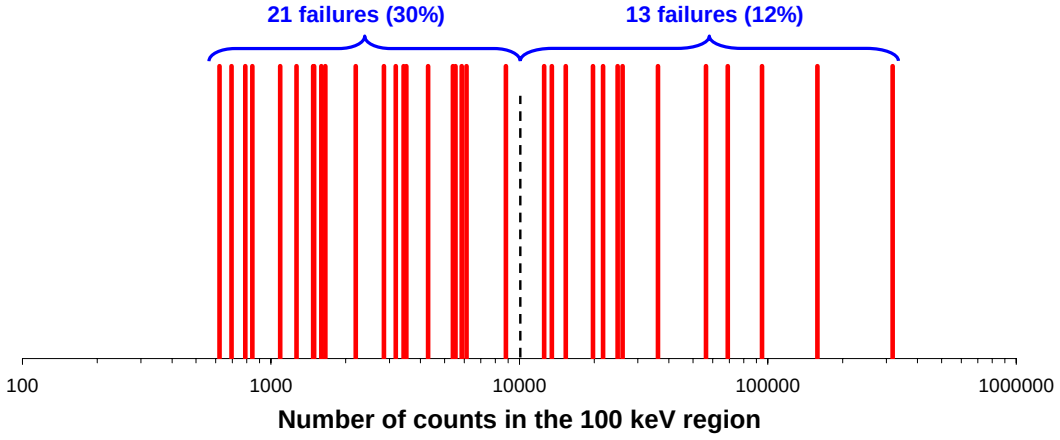


**Fig. 4** Failures distribution for MGA ver.9.63H

The main reasons, when MGA ver.9.63H failed analysis, were either low counting statistics or presence of a high background. MGA ver.9.63H runs internal tests and, if the counting statistics is too low, or the peak-to-background ratio in the 100 keV region is above a certain value, it does not report any results. MGA ver.10, in contrast, has more relaxed internal requirements for waste and unusual spectra, and in our study it was able to report isotopic compositions for each spectrum. Fig.5 shows an example of MGA peak fits in the 100 keV region for difficult to analyze spectrum. This spectrum has both low counting statistics and high peak-to-background ratio. As a result, MGA ver.9.63H failed to produce any report, and MGA ver.10 reported $^{240}Pu_{eff}$ weight fraction with RSD of about 30%. Note that the yellow peak in the spectrum corresponding to 104 keV of $^{240}Pu$ overlaps with few other peaks and is just barely above the background.

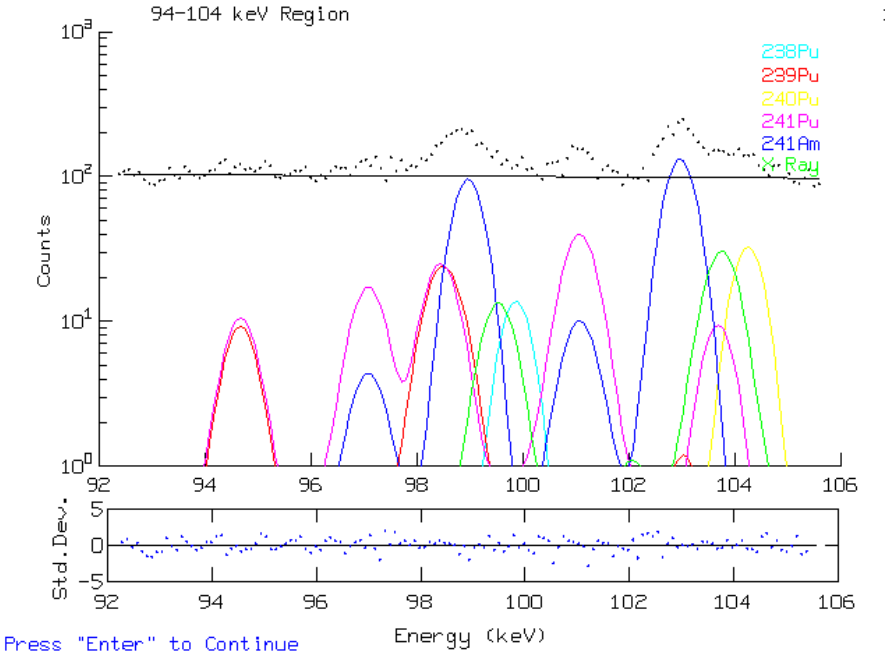


**Fig. 5** MGA ver.10 fits for the 100 keV region.

## 4. Conclusion

With the standard plutonium samples available we have taken a practical approach to directly assess the performance limit of MGA for the determination of $^{240}Pu_{eff}$. Two versions of the code, MGA ver.9.63H and MGA ver.10, were used and compared. The result showed that, when the analyzed spectrum has 7000 – 8000 counts in the 100 keV region, both codes can (usually) *reliably* determine a weight fraction of $^{240}Pu_{eff}$. In some cases, presence of a high background or low counting statistics prevented MGA ver.9.63H from reporting results. This is because this widely used version of MGA has strict internal requirements that could not be met.

Knowing the isotopics code performance limit can help in setting up systems and in optimizing the counting geometry. For an unknown case, for example, the count rate may be predicted by using the ISOCS efficiency estimating tool seeded with the decay data for the various Pu isotopes. ISOCS [5, 6] allows one to create an absolute efficiency curve for the 100 keV region, and thus the count rate in that region may be estimated. If the predicted rate for the measurement configuration considered, which may be a waste container, say, results in a count that exceeds the number required in the data acquisition period allowed we can have confidence that the isotopic code will run and yield reliable results. The database of spectra required in this work can also serve as an empirical test bed in the sense that the experiment closest to the predictable rates can be used as the gauge.

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[2] J. Morel, B. Chauvenet, *Intercomparaison des Measures de Composition Isotopique du Plutonium Par Spectrometrie X et Gamma Resultats de L'action "PIDIE" Rapport Final,* Rapport CEA-R-5582, Centre d'Etudes Nucleaires de Saclay, 1992.

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# ATOM History

- UKAEA had multiple incompatible systems and processes for NMA
  - CNMAS at Dounreay
  - SMART at Windscale
  - ANUMAP at Harwell and Winfrith
- By the mid-1990s they all needed replacing to meet the evolving safeguards (and related) legislation, criticality control, stockpile management, audit, IRRs, NISR and liabilities tracking and management
- In 1996, UKAEA decided to create something future-proof - ATOM
- Over the next 5 years >£5 million was invested in ATOM
- ATOM became operational at all sites in 2001

# The Current Situation – A Mismatch

- What is needed?
  - New Reporting Regulations come into force in 2008
  - Accommodate a change in regulatory approach by DG-TrEn
- What is there today? A typical NMA application is:
  - Home made and > 10 years old
  - No longer supported by the people who wrote it and of limited functionality
  - Running on obsolete hardware and/or operating systems
  - Used differently by different sites within the same organisation
  - Limited to identification rather than tracking and audit
  - Not designed with integrated DG-TrEn reporting

# How should I resolve the mismatch? Choose…

- **In-house and re-develop**
  - Can I afford the cost? Can I afford the time?
  - Do I have SQEP'd staff and how do I retain them?
  - How do my NMA skilled staff fit this in with their day jobs?
  - How do I provide ongoing support?
  - How can I be sure that the regulators will give me approval?
- **In-house and new build**
  - All the above issues and won't that take even longer and cost even more?
- **Use a third party developer**
  - I have to write the spec; if I get it wrong it's going to cost me!
- **Buy a package**
  - Is there one that meets my NMA needs AND conforms to
  - DG-TrEn reporting regulations AND minimises my risk?
- **Subscribe to a Service**
  - ?





UKAEA Restoring our Environment

FUJITSU

# The Solution

***"The ATOM service is the most effective, least cost and lowest risk way to assure regulatory compliance and avoid nuclear material tracking issues"***

- Most effective regulatory compliance solution
- Most effective Nuclear Material tracking solutions
- Least cost solution
- Lowest risk solution





UKAEA Restoring our Environment

FUJITSU

# Most effective regulatory compliance solution

- ATOM is endorsed by the OCNS, DG-TrEn and the NII
- Conforms to all current and anticipated DG-TrEn reporting requirements
  - Rigorous process, system design and audits comply with  DG-TrEn specc
  - Independent system audit (private) to measure/assure compliance
  - Already fully specified to meet the new 2008 regulations
- Fully supported by acknowledged regulatory experts from the UKAEA





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# ATOM – Lowest Cost Solution

- Zero CAPEX
- Minimum application development and support costs
  - Shared across the user base
- Elimination of duplicated costs
  - Single integrated system across the organisation
- Minimum system service costs
  - Fujitsu S&R approach to drive down cost of support
- Reducing costs over time
  - Prices fall as subscriber base grows





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# ATOM – Lowest Risk Solution

- Ahead of the game
  - Will beat the deadlines
- Can be piloted today
  - ATOM available now
- Proven in practice
  - In use today at UKAEA sites
- Delivered by a reliable partnership
  - Service delivered through global IT leader Fujitsu
  - UKAEA nuclear materials handling knowledge and experience
  - A professionally developed and tested software solution, standards based (including BSI TickIT Guide, ISO9000 etc) with documentation, version control and software support





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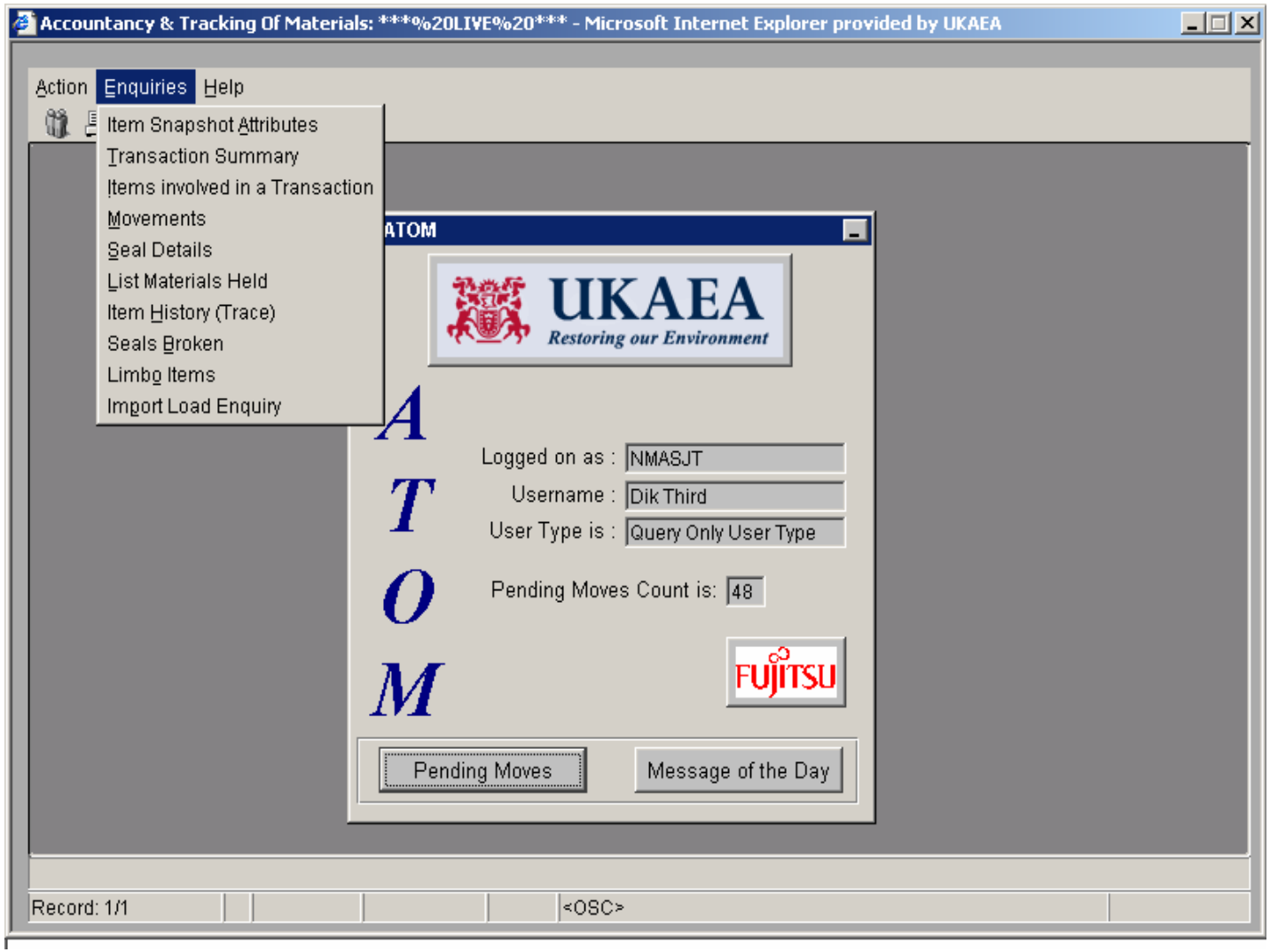
# ATOM Enquiries Menu





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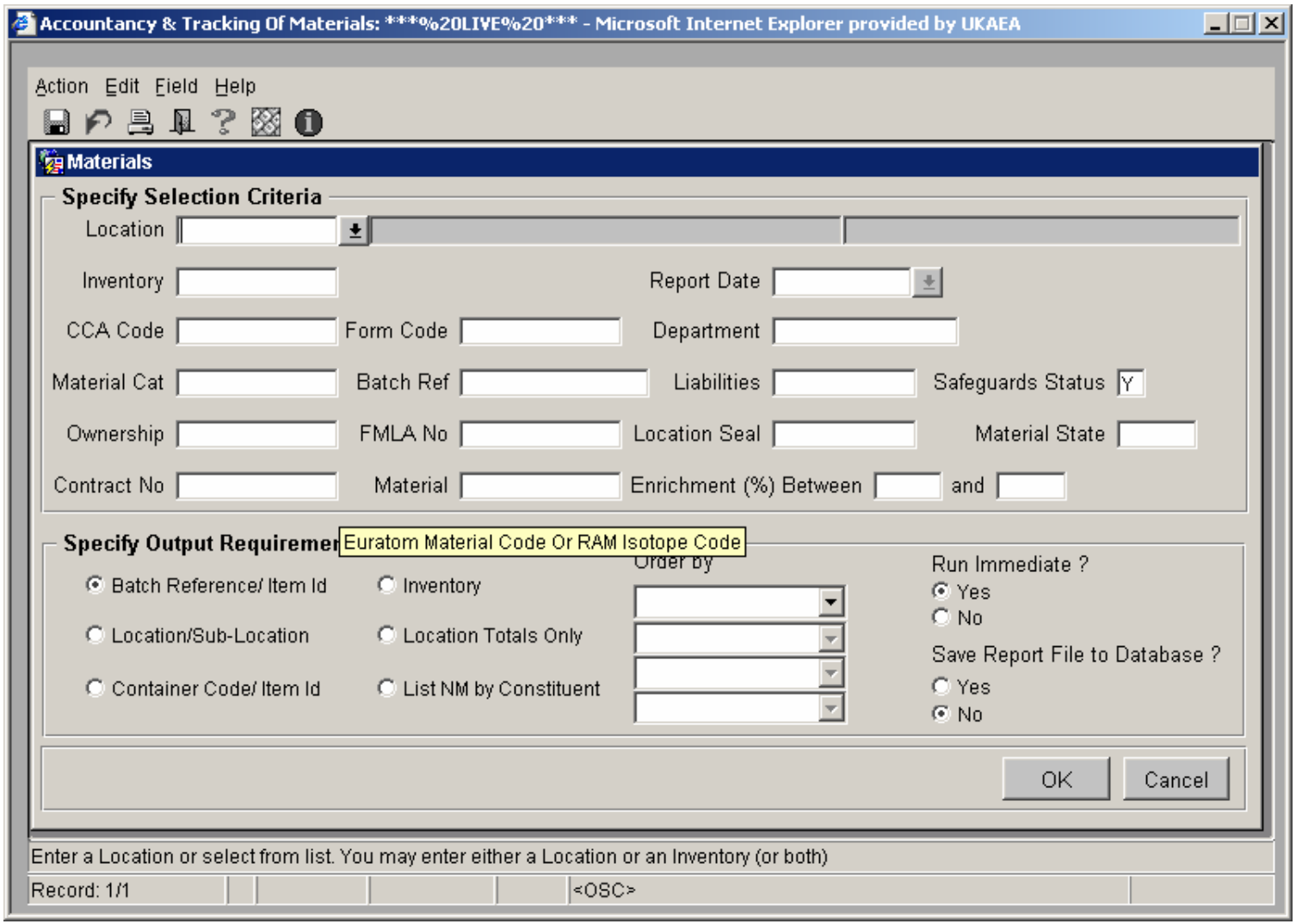
# ATOM List of Materials





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# What Should I Do Next?

The ATOM service is the most effective, least cost, and lowest risk way to assure regulatory compliance and avoid nuclear material tracking issues

+

The ATOM service is up and running today

=

You should take the necessary steps to prove to yourself that the ATOM solution is right for you

You should commence a pilot to experience ATOM in practice and engage with the ATOM team to answer all your other questions about the service





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# Plutonium mass measurements in waste drum by neutron coincidence counting: principle and recent improvements

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***Abstract:***

*In the framework of the enforcement of the French domestic safeguard regulations which are incumbent on the minister in charge of industry, the Institut de radioprotection et de sûreté nucléaire (IRSN) contributes by notably producing measurements on nuclear materials. The present paper is focusing on the measurement of plutonium quantities, and more particularly the plutonium contained in waste drums produced by various operators.*
*One of the objectives of IRSN's contribution in the field of domestic-safeguard-related inspections is to produce a reference value and a reference uncertainty of the quantity of plutonium contained by the drum. The estimation needs to be performed on site and directly on the operator's samples. In addition, the measurement needs to be performed by IRSN's own means, that is to say independent from the measurements means of the operator. During the inspections, once the reference values are produced by IRSN's measurements on several samples, these values are compared with the operator's declared values, and thus permits to derive a conclusion on the quality of the operator's nuclear material management system.*
*On a technical basis, this paper presents a passive neutron coincidence counting device named FUNE which is used for measuring quantity of plutonium in waste drums. This system is designed to be transportable; it is installed and operational on site in less than one hour. This paper will recall first the principle of measurement of this device, its main components and the different tests that have been conducted for the qualification of the system. In second, it addresses the recent improvements that have been achieved on the data acquisition and treatment system and on the methodology: a new electronic device enables a direct counting of the pulses produced on every single $^{3}He$ neutron detectors, giving some information likely to be used to localize the radial position of the plutonium potential sources.*

## 1. Introduction

The determination of quantities of plutonium in waste drums is one of the important focuses of attention established by the French domestic safeguards authority, and orienting the inspection effort. In this framework, the device FUNE (figure 1) based on passive neutron coincidence counter was developed and elaborated at IRSN. The technique is based on a method described further in the paper.
The constraints for on site use have led the designers to think of a device which is transportable (figure 2), modular (figure 3), geometrically adapted to any type of drum and container, fast and easy to implement.
Moreover, the measurement time per drum being limited, the device must also present high performances for measuring a maximum of drums during a campaign.

This paper describes all the steps leading to the determination and the optimization of the device parameters such as efficiency and setup up calibrations curves using certified plutonium sources.
Regarding the accuracy of the measurement produced by the device, three causes of errors on the final result have been identified and taken into account [1], [2]: (a) neutrons background effects, which depends on the environment; (b) the effect of sources positioning in the drum; (c) neutrons absorption effects, due to the filling up matrix.
The materials used in the qualification process are plutonium oxide sources with masses ranging from 50 mg to 15 g of plutonium. The standard matrix is a vinyl matrix (figure 4), designed to simulate four different densities (about 0,1 g.cm $^{-3}$ 0,2 g.cm $^{-3}$ 0,3 g.cm $^{-3}$ 0,4 g.cm $^{-3}$ ). All these materials are meant to simulate with the best accuracy actual waste drums that are to be characterized during inspections. In all this study, the isotopic composition is supposed to be determined with an accuracy unlikely to induce significant errors on the final result. Thus the question of isotopic composition determination is not addressed in this paper.

- Figure 1- FUNE, full experimental set up

- Figure 3 - Parts of the set up once disassembled and ready for transport

- Figure 2 - FUNE modular neutron detector blocks

- Figure 4 - Vinyl matrix used for the FUNE calibration

This paper describes also the study in progress with the aim to reduce the uncertainties due to the positioning effects of materials in the drum. These improvements are obtained with using a so called neutron imaging technique (NIT) method.

## 2. Description of the experimental set-up

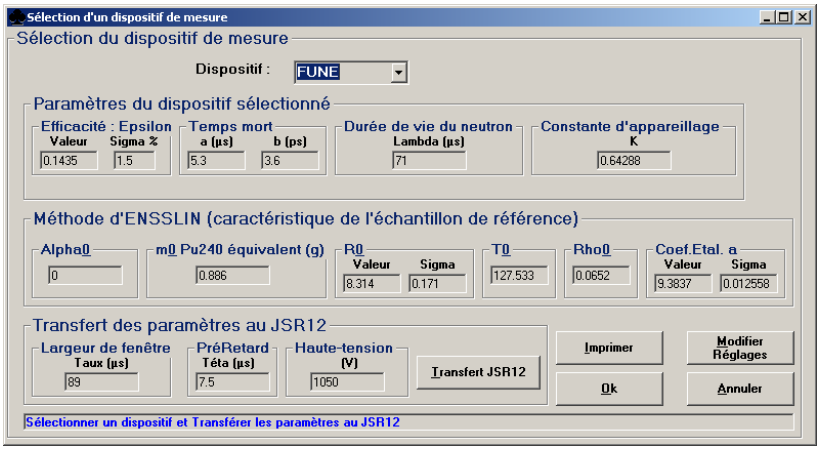
The cell consists in fourteen polyethylene blocks into which twenty eight Helium-3 detectors of type XERAM 150NH100C have been introduced (2 detectors per detection block). Fourteen charge amplifiers called ACHSI95 and developed by EURISYS-MESURES Company, allow the charges collection and the discrimination of neutron impulse from photon impulses.

In order to neutralize the differences of spectra between the twenty eight Helium-3 detectors, the well was equipped with a revolving plate. Two mobile doors attached by hinges give an access to the cavity of measurement so that the waste drum to be analyzed can be introduced. Two reflectors of polyethylene have been added at the top and the bottom of the cell to improve the efficiency and minimize the neutrons losses.
An electronic acquisition and treatment system, named JSR-12 and developed by CANBERRA Company, is used for the treatment of the impulse sequence. The software which is called ATENE has been developed by IRSN and is connected to the JSR-12. The software allows calculating the final quantity of plutonium. Moreover, an electronic module developed by DANELEC Company allows switching easily from a global counting – full cell -- to a transmission measurement – half cell (Figures 5 and 6).

- Figure 5 -


- Figure 6 -

The characteristics of FUNE and the shift register parameters are read and transmitted to the JSR-12 by the ATENE software (figure 7).




- Figure 7 -

## 3. Principle of the measurement

### 3.1. Calibration principle

The method consists in recording the spontaneous neutron fissions produced in the disintegration of the even isotopes of plutonium. The sorting of the neutron impacts in detectors corresponding to spontaneous fissions from the neutrons due for instance to ($\alpha$,n) reactions is obtained by use of a shift register technique [3].

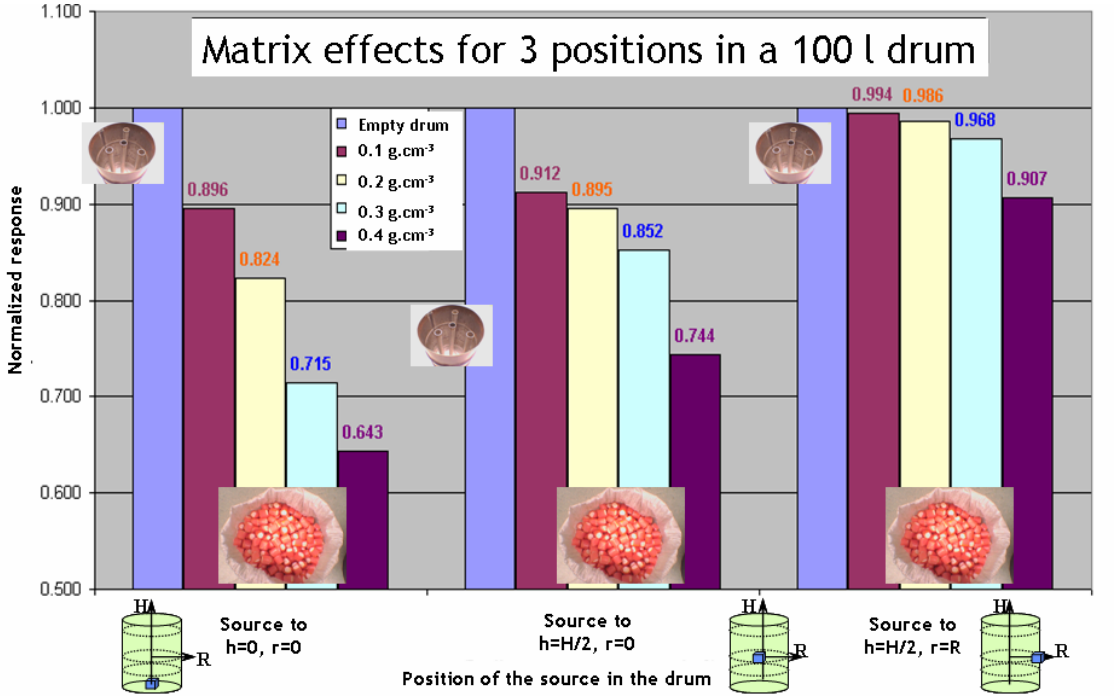
Based on this technique, calibration curves have been determined to relate the neutron coincidence counting to the equivalent mass of plutonium 240, with respect to the absorption of the matrix of the drum. For that purpose, a correction factor is applied which is deduced from a transmission measurement using an external californium 252 source.

Therefore, a preliminary work of qualification consisted in establishing, for each matrix of density $\rho_i$, a calibration curve using certified plutonium sources.
As a first estimate, the data are correlated following the following linear relationship:

$$m^{240}Pu_{eq} = a_{\rho_i} . R_{\rho_i} + b_{\rho_i}$$

$a_{\rho_i}, b_{\rho_i}$ are the coefficients of the linear regression relative to the calibration curve obtained for a matrix of density $\rho_i$. $R_{\rho_i}$ is a coefficient which corresponds to the doublets counting rate for a drum with a matrix

density equal to $\rho_i$. The equation above supposes a homogeneous matrix and no induced fission. Figure 8 shows the influence of the matrix effects considering three different positions for the source in the drum.




- Figure 8 -

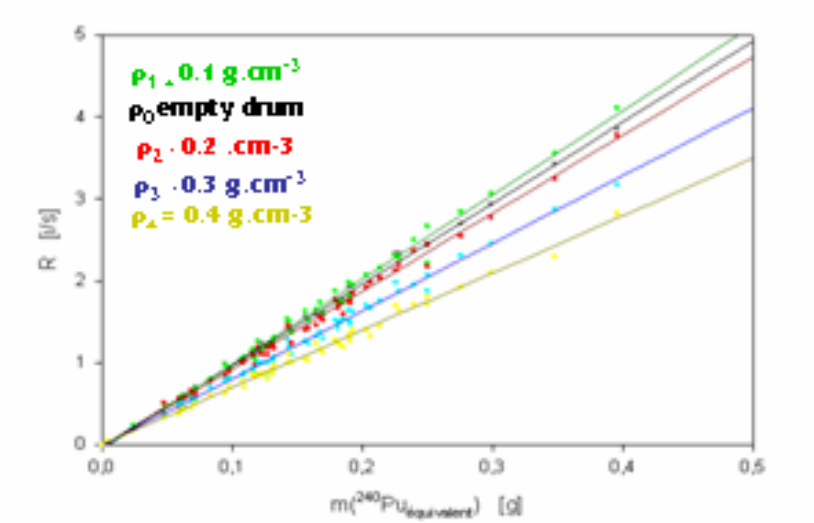
If the neutron attenuation phenomenon is taken into account, the predicted signal is reduced by more than 35 % (see figure 8 for a source located on h=0, r=0 and with a density $\rho$ =0.4 g.cm$^{-3}$).

In the following, the quantification of plutonium can be performed by applying one of the two following methods:

- An abacuses calibration method with one curve or one equation per matrix
- A surfaces calibration method in which all the data is covered by only one equation

### 3.1.1 Classical calibrations

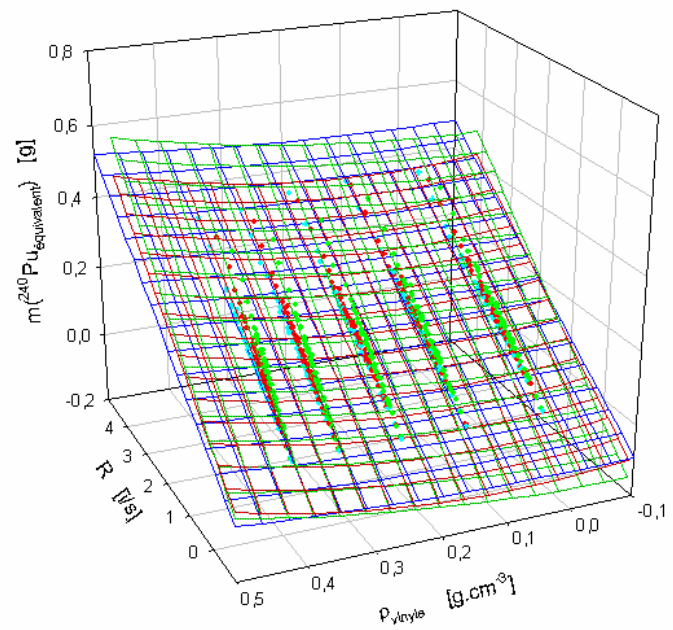
With the abacuses method, after determining the matrix density, the method consists in choosing between five calibration curves (figure 9). A simple method consists in assuming that the appropriate curve is the one obtained with the density being the closest to the measured density for the matrix in the drum. Each curve has been fitted based on fifty measure points.




- Figure 9 - Calibration curves obtained for various matrix density values.

### 3.1.2 Surfaces calibration

Another way to calibrate the approach is to consider only one equation relating the equivalent 240-plutonium mass to the double counting rate: $m^{240}Pu_{eq} = a_{\rho_i}.R_{\rho_i} + b_{\rho_i}$ . The relation can be represented by a surface (figure 10). This method is very close to the previous one in principle, only the interpolation method is changed. The accuracy is comparable, say better, in many cases, nevertheless the surface method can lead to unsatisfactory results when one approach the limits of range. In that last case the use of the abacuses method should be preferred.




- Figure 10 - Calibration surface interpolated from various-matrix-density-based calibration curves.

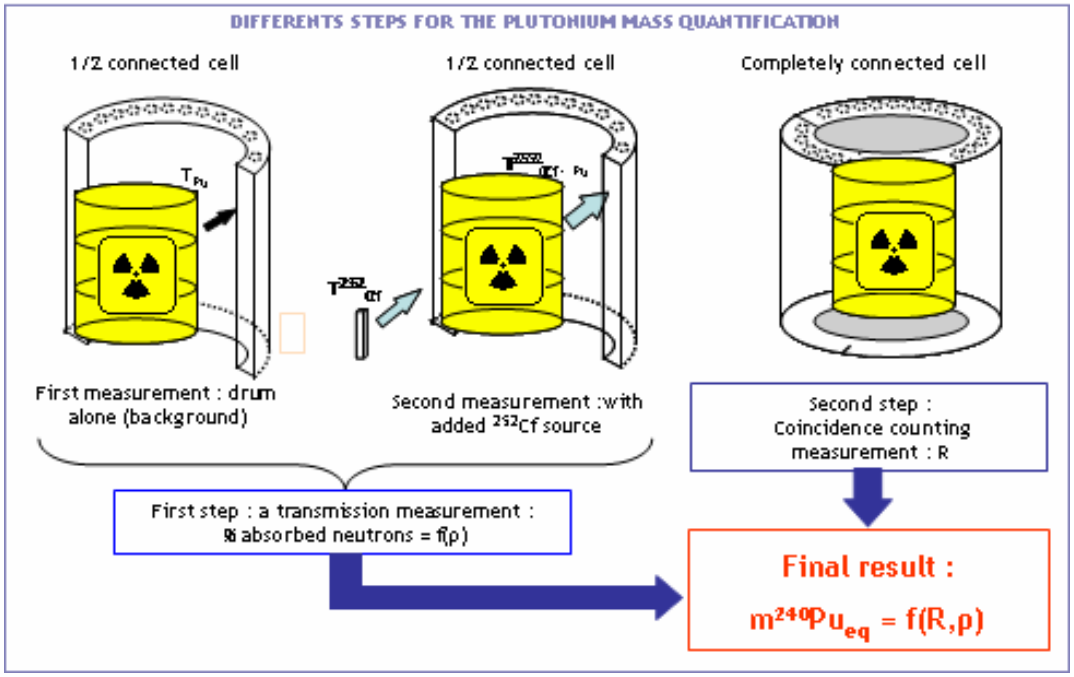
## 3.2. Determination of the matrix density

On site, the exact content of a drum (metal, wood, concrete, vinyl, etc.) is never directly accessible to the inspectors, yet a simple weighing can lead to a good estimate of the density. However this value does not indicate the real correction factor to be applied due to neutron absorption. Indeed, neutron absorption depends on the nature of the matrix and not only on the density. For example a matrix constituted of light elements is inducing significant neutron absorption in comparison to the absorption induced by a matrix constituted with heavy elements. In these conditions, how could one estimate a density $\rho_i$ likely to be used to characterize the neutron absorption?

To answer that question, several matrixes of vinyl type have been prepared at IRSN, with a density ranging from 0.1 to 0.4 g/cm$^3$. A measurement of the neutron flux emitted by a Californium-252 source passing

through the matrix has been performed, also known as a transmission measurement. This allows evaluating the neutron absorption with respect to every matrix tested. With this method, the neutron absorption correction factor is determined experimentally, independently of the nature of the matrix and leads to an equation of the type: $\rho_i$ = *f(% absorbed neutrons)*.

Finally, including the background estimation and the neutron absorption due to the matrix determination, three measurements are necessary to proceed to the quantification of the plutonium in a waste drum (figure 11).




- Figure 11 -

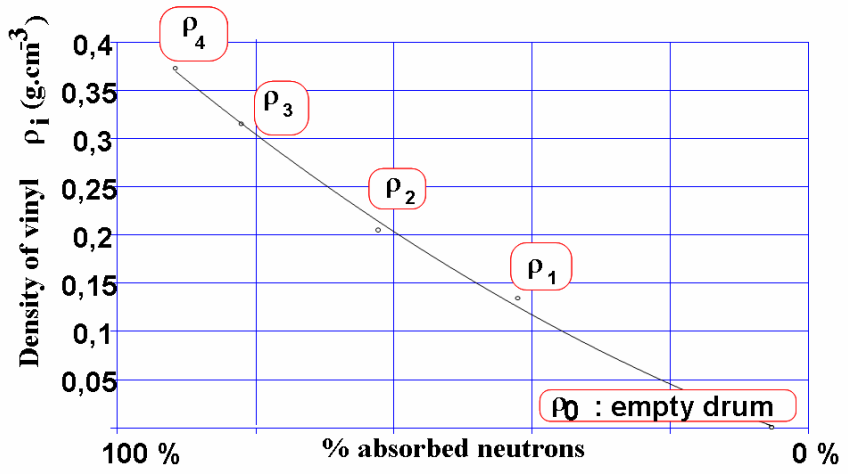
For the first two measurements, only half of the detectors of the chamber are used (figure 11).

**First measurement.** The drum is alone and placed into the half-cell; this sequence intends to measure the background and the measurement result is noted $T_{Pu.}$

**Second measurement.** This is a transmission measure using the Californium-252 source. The drum is put between the half-cell of detectors and the source of californium. The result, noted $T_{252Cf+Pu}$, integrates the contribution of neutrons emitted from the californium 252 source and those resulting from the plutonium contained in the waste drum. The percentage of absorbed neutrons is obtained with the following formula, and uses the known neutron flux emitted by the Californium-252 source $T_{252Cf}$:

*% absorbed neutrons* = $[T_{252Cf} - (T_{252Cf+Pu} - T_{Pu})] \times 100] / T_{252Cf}$.

This approach is independent of the intensity of the $^{252}$Cf source. After that, one uses a calibration curve produced by IRSN (figure 12) relating the percentage of absorbed neutrons to the density of vinyl.




- Figure 12 -

In this study, for an unknown drum, the nature of the matrix is not important since the measured value is a percentage of absorbed neutrons. So it becomes possible to apply the curve shown on figure 12 for any kind of matrix (metal, wood, concrete, etc) likely to be contained by unknown drums.

**Third measurement.** The drum is measured alone, the cell is complete and fully connected. In this case the measurement corresponds to the actual coincidence counting R.

So, knowing R and $\rho$, a calibration of type $m^{240}Pu_{eq} = a_{\rho_i} . R_{\rho_i} + b_{\rho_i}$ is applied (see 3.1) to deduce the final result.

The range of applicability of this method is defined for $\rho$ being always comprised between 0 and 0.4 g/cm$^3$.

### 3.3. Some results

IRSN participated in EQRAIN n°3 and n°4 round robin exercises, organized by CEA/CETAMA for the characterization of several plutonium waste drums B, C, D and A', B', C' drums were specifically constituted for this exercise. They were filled with a combustible matrix of a given density (0.17g/cm$^3$) and various plutonium point sources, a priori. Cesium-137 has been added to the contents of B, C and A', B' drums. D is an actual drum with a density of approximately 0.3 g/cm$^3$.
Table 1 shows the results obtained on these drums, using the FUNE device and the approach described above. There is no reference value for the real D drum. The value of the standard deviation for FUNE is ±15%.

| N° | Matrix type - density | Content | CETAMA Reference plutonium mass (mg) | Measurement using FUNE | | |
|---|---|---|---|---|---|---|
| | | | | Pu mass (mg) | ± | σ (mg) |
| B | Combustible - 0.17 g/cm$^3$ | Pu + $^{137}$Cs | 76 | 65 | ± | 10 |
| C | Combustible - 0.17 g/cm$^3$ | Pu + $^{137}$Cs | 204.6 | 204 | ± | 31 |
| D | Combustible - ~0.3 g/cm$^3$ | real drum | No value | 86 | ± | 13 |
| A' | Combustible - 0.17 g/cm$^3$ | Pu + $^{137}$Cs + $^{154}$Eu + $^{134}$Cs | 126 | 108 | ± | 16 |
| B' | Combustible - 0.17 g/cm$^3$ | Pu + $^{137}$Cs | 124 | 111 | ± | 17 |
| C' | Metallic type - ~0.25 g/cm$^3$ | Pu | 173 | 163 | ± | 24 |

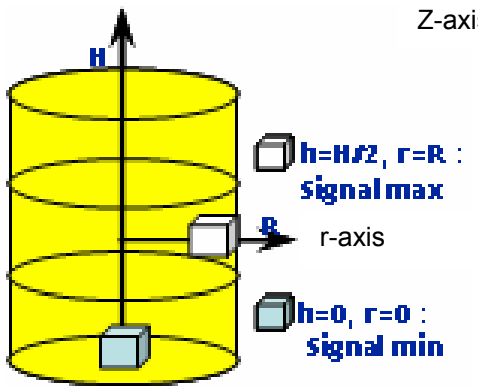
- Table 1 -

## 4. Improvements and perspectives with FUNE: localization of neutron sources

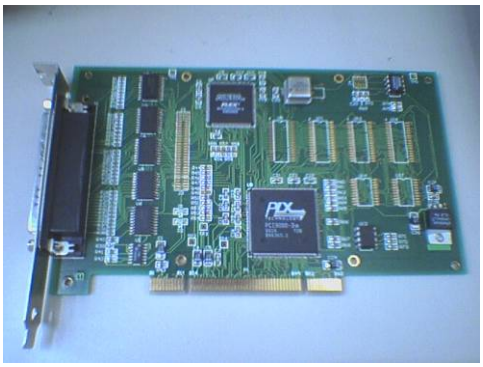
The uncertainties of a measurement performed with the FUNE device are influenced by:

- the matrix
- the coefficients of the linear regression involved in the calibration process
- the statistic counting precision
- the isotopic composition more specifically related to the even plutonium isotopes
- the position of the sources in the drum

Among these uncertainties, the principal source of uncertainty is due to the poor information about the localization of the sources in the drum. A study showed that the localization corresponding to the most unfavourable positions in the drum are those indicated on the figure 13. The ratio of uncertainty due to the localization effect can reach 75% of the total uncertainty.




- Figure 13 -



- Figure 14 - MEDAS card

A modification of the FUNE electronic circuits has been achieved in order to measure the neutron counting of each of the fourteen detection blocks. This possibility was offered with the acquisition of a multi scale counting card, so-called MEDAS (Multi Events DAtation System) developed by CESIGMA (figure 14) company: this card is a PCI card inserted in the PC to record each counting rate coming from each detection block.

The aim of the study is to investigate if it is feasible to use the total neutron count rate in order to have an indication on the localization of the neutron source term and subsequently to take this information into account in order to reduce significantly the total uncertainty of the measurement. With the current detection geometry of the FUNE device, there is only radial localization (r-axis on Figure 13) and no axial localization (z-axis on Figure 13). Subsequently, if the result of the study is promising, a new design of the FUNE device would be necessary in order to measure also the axial localization.

Several methods have been investigated:

- Emission computed tomography using the Algebraic Reconstruction Techniques (ART), additive and multiplicative [4];
- Emission computed tomography using the Expectation Maximization (EM) method [4];
- Neutron Imaging Technique (NIT) [5].

For each method, both MCNP modelling and experimental measurements have been carried out using californium 252 and plutonium sources.

A mathematical relation has been established to take into account the position of the sources, determined by these methods, in the calculation of the plutonium mass: it relates the counting obtained for an unspecified position of the source to the counting obtained if this same source is placed in the centre of the FUNE device. With this relationship, it is possible to use the existing calibration of the FUNE device for the calculation of the plutonium mass because the calibration has been carried out by centring the sources in the cell.

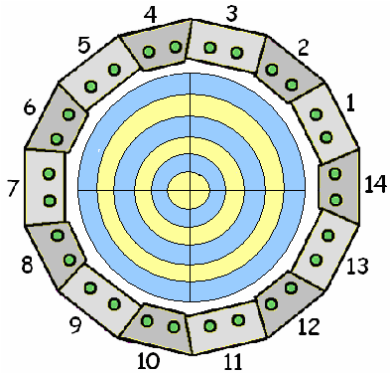
Emission computed tomography using ART and EM algorithm

These methods give good results when it is used with data simulated by MCNP. However, it is unsuited to the localization of sources when dealing with experimental data. The principal obstacle is due to the passive character of the method: the useful signal is too weak to be treated by the reconstruction algorithm. The sources placed in the internal mesh, in the centre of the device or with the intersection of several meshes generate a too weak useful signal for the treatment by ART or EM algorithm, and are thus localised in an approximate way.

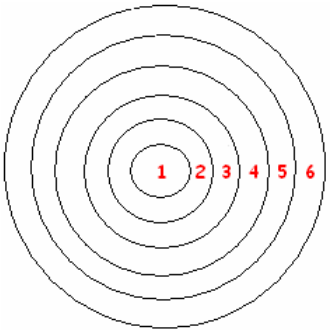
Neutron Imaging Technique

The methodology that is used is described in the work done by Wilkins et al. [5]. The drum is divided into elementary annular sectors in which the fissile material could be localized. The Neutron Imaging Technique is based on experimental measurements (calibration measurements), using a neutron source and several homogenous matrices, in order to obtain a single figure $\overline{\langle\sigma\rangle}_k$ that is different when the known neutron source is moved from one sector to the adjacent one. The same figure is calculated for the unknown drum and the matching with the calibration measurements enables the localization of the unknown source.

The grid that has been defined for the present study is presented figures 15 and 16.




- Figure 15-




- Figure 16 - annular sectors

To apply that technique, the size of the grid needs to be optimized: if the number of sectors is increased, the difference between two adjacent values of $\overline{\langle\sigma\rangle}_k$ decreases and may become too small to decide where the source is located. In the case of the FUNE device, optimisation calculation leads to define 6 values of $\overline{\langle\sigma\rangle}_k$ as shown in figure 16.

Definition of $\overline{\langle\sigma\rangle}_k$

The following procedure is done for various matrix density values, an empty drum, 0.1, 0.2, 0.3, and 0.4 matrix density. For each matrix density, a set of 6 values of $\overline{\langle\sigma\rangle}_k$ is obtained.

A neutron source is set successively in the six annular sectors and four measurements are done, doing between each measurement a 90° rotation of the drum. The number of rotation is a parameter that needs also to be optimized. More rotation gives more accurate results but longer acquisition. For the FUNE device, at least 2 rotations are needed. With respect to the symmetry of the cell, four rotations seem to be a good compromise between the required accuracy and the counting time.

4 rotation leads to six 4X14 matrix of $\langle R_{ij}\rangle$ which is the normalized total counting rates for the $j^{th}$ detector block and the $i^{th}$ rotation.

For each detection block (1 to 14) the following parameter is calculated: $\langle\sigma_j\rangle = \sqrt{\frac{n\left(\sum_j \langle R_{ij}\rangle^2\right) - \left(\sum_j \langle R_{ij}\rangle\right)^2}{n(n-1)}}$

where:

- n : number of rotations (i.e. equal to 4).
- j : detector block (1 to 14)
- i : drum orientation (1 to 4).

Eventually, the mean value is calculated. $\overline{\langle\sigma\rangle}_k = \frac{\sum_j \langle\sigma_j\rangle}{14}$

To understand the meaning of this parameter lets us consider a source placed at the centre of the sector (area 1 in figure 16). The $\langle R_{ij}\rangle$ values for all the block will be roughly equal and the $\overline{\langle\sigma\rangle}_k$ value will be very close to zero. Conversely, if the source is placed far from the centre of the sector (area 6 in figure 19), the $\langle R_{ij}\rangle$ values differs from one to the other and $\overline{\langle\sigma\rangle}_k$ increases with the distance to the centre of the sector.

For an unknown drum of a given matrix, the same procedure is followed and the calculated value of $\overline{\langle\sigma\rangle}_k$ gives the localization of the neutron source in the drum with regard to the grid defined in Figure 16.

The measurement procedure is the following:

- Perform a transmission measurement to evaluate the matrix density for the unknown drum;
- Choose the correct reference matrix with respect to the previously measured value.
- Perform 4 measurements with a 90° rotation of the drum and calculate $\overline{\langle\sigma\rangle}_{exp}$.
- Compare $\overline{\langle\sigma\rangle}_{exp}$ with the six $\overline{\langle\sigma\rangle}_k$ different values. The closest value indicates the position of the source between the six possible positions on the grid.

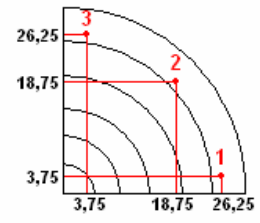
Measurements with a $^{252}$Cf neutron source have been done to estimate the $\overline{\langle\sigma\rangle}_k$ values in two cases corresponding respectively to an empty drum and a drum filled with a 0.3 g.cm$^{-3}$ dense matrix.

| | $\overline{\langle\sigma\rangle}_1$ | $\overline{\langle\sigma\rangle}_2$ | $\overline{\langle\sigma\rangle}_3$ | $\overline{\langle\sigma\rangle}_4$ | $\overline{\langle\sigma\rangle}_5$ | $\overline{\langle\sigma\rangle}_6$ |
|---|---|---|---|---|---|---|
| empty drum | 0.018 | 0.043 | 0.067 | 0.084 | 0.115 | 0.131 |
| 0.3 g.cm$^{-3}$ matrix density | 0.025 | 0.077 | 0.126 | 0.169 | Not measured | Not measured |

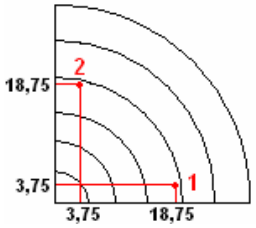
- Table 2 -

These results show that the $\overline{\langle\sigma\rangle}_k$ parameter is sensitive enough to localize a source in a particular area of a drum for a given matrix value.

Several measurements with different source localization have been done to test this method using the drum with the 0.3 g.cm$^{-3}$ matrix density. Figure 17 and 18 illustrates some of the results.




Test 1
- Figure 17 -




Test 2
- Figure 18 -

The results were the following.

| Test 1 | Position 1 | Position 2 | Position 3 |
|---|---|---|---|
| $\overline{\langle\sigma\rangle}_{exp}$ | 0.124 | 0.124 | 0.125 |

| Test 2 | Position 1 | Position 2 |
|---|---|---|
| $\overline{\langle\sigma\rangle}_{exp}$ | 0.092 | 0.092 |

Using the results of table 2, one can see that the localization of the source in sector 6 for the first test is confirmed as well as the localization of the source in sector 4 for the second test.

## 5 Conclusions

The qualification of FUNE was the subject of two studies.
On the one hand the determination of a method that takes into account the major sources of uncertainties concerning the quantification of plutonium in a waste drum. In particular, a transmission measurement using a californium source allows minimizing the error due to the matrix absorption for an unknown drum. On the

other hand, among sources of uncertainties in a waste drum, the poor knowledge about the source position can lead to an important bias in the quantities of plutonium estimated. By now, this error has been estimated from the standard deviation of a measurement result being influenced by various parameters.
The feasibility of Neutron Imaging Technique has been demonstrated with the FUNE device for a 2-D localization of a neutron source. With this method, the drum is divided into elementary annular sectors in which the fissile material could be localized. For the FUNE device, six elementary annular sectors have been defined from the centre to the circumference of a drum. The counting rate per detection block is recorded for each rotation, using four different angular positions. The results show that the figure measured to characterize the localization of the neutron source is sensitive enough to localize it in a particular area of a drum for a given matrix value.
To implement this method in the FUNE device, more calibration is needed. Moreover addition, a new design of the device is required for three-dimensional localization capabilities.

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# A compact PC Cluster for Monte Carlo Simulations at JRC-Ispra.

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***Abstract:***

*At the JRC Ispra, Monte Carlo simulation programs like MCNP and MCNP-PTA are becoming more and more a fundamental tool in the design of new neutron coincidence counters and neutron multiplicity counters. These programs also provide an alternative way for the calibration of these neutron counters for the assay of new sample materials, without the need of new nuclear reference materials. A new PC Cluster has been build to satisfy the increasing demand of computing power, which had the following design requirements: high speed processors, large memory size per node, compact dimensions, high reliability, easy to use, and provide ways for future expansions/upgrades. This paper reports on the design, realisation and performances of the new cluster. Also discussed are the choice of the Operating System (Linux) and various aspects of the software required for running MCNP and MCNP-PTA in parallel with Parallel Virtual Machine (PVM).*

Keywords: PC cluster, PVM, MCNP, MCNP-PTA

## 1. Introduction

Computers are continuously evolving, and this paper can only give a snapshot of possible computing solutions for a small group of Monte Carlo users. Nevertheless, we can assume that the general ideas and trends will still be valid for at least one or two of Moore's laws cycles (of 24 months).

The PC cluster described in this paper is really optimised for the particular case of Monte Carlo calculations in safeguard applications, which because of their nature are quite easy to execute in parallel. For example, imagine running a 1 million histories MCNP calculation on 1 million computer nodes: starting a single particle history on each node with a different starting random number, doing the radiation transport on each node independently and at the end collecting the tallies from all nodes to calculate the average result. Besides ray tracing in 3D imaging, there are very few other computing problems that can be executed in parallel that easy. Normally calculations are interdependent, and require both sophisticated partitioning of the calculation and high bandwidth data exchange, in order to profit from multiple processors. One exception is criticality calculations with MCNP, where numerous iterations are necessary and the demand on the data exchange saturates the performance increase with the number of nodes.

The open source Parallel Virtual Machine (PVM) software library is used for the parallel execution of the codes. The reason for not using the other well known library, called Message Passing Interface (MPI), is mainly historical: i.e. in 1993 MCNP (version 4a) featured for the first time distributed processor multitasking the PVM library was used. Although, MPI may increase performance in some cases, the results with PVM are already very good.

When running MCNP, with PVM or MPI, the Monte Carlo simulation is split in a single master task (on a server) and many micro-tasks (on the cluster nodes). After a cluster node finishes such a micro-tasks, it will send a message with calculation results to the server. After reception of all messages, the master task calculates intermediate results and re-organises the work load for the next rendez-vous. By optimising the number of rendez-vous, the bandwidth requirements of the interconnecting network can be reduced without losses in computing performance of the cluster. MCNP-PTA and PTA codes have also been adapted to run in parallel with PVM.

The Gigabit Ethernet standard (GbLAN) provides high bandwidth (1000 Mbps) communication via twisted wire cabling (UTP) with a maximum length of 100m. It is used in 43% of today's supercomputers, which are based for 72% on cluster architecture [1]. Other networking solutions, like Myrinet and Infiniband, have better performances but are also much more expensive, while there are plenty of very economic computer motherboards with an embedded GbLAN interface. This paves the way to a high performance computing solution using inexpensive personal computer hardware: a Beowulf cluster. What follows is a discussion of some aspects of the Compact Beowulf Cluster System (CBCS), which has been used for the JRC Ispra cluster (4 nodes) and for a 16 node cluster installed at the Safeguards Technical Support Unit of the IAEA in Vienna.

## 2. Processors

The choice of a processor is mainly based on: performance, power dissipation, processor cost, system cost and system cooling requirements (see Tables 1 and 2).

**Table 1.** Performance characteristics of AMD and Intel processors: processor clock (in Ghz), Thermal Design Power (TDP, in Watt), and number of cores per processor chip. Configurations of a 16 core system are given under the System header: number of required motherboards and processor chips. The performance values rates (SPECfp_rate2000, SIXTRACK VALUE ONLY) by the SPEC organisation [2] for each processor are also listed (values in standard, not bold, font are calculated by interpolation based on clock speed). The theoretical performance of the 16 core system is given in the last columns (absolute and percentage with respect to the AMD Athlon64 3800+).

| | clock | TDP | | System | | SPECfp_rate2000 | | | Performance | |
|---|---|---|---|---|---|---|---|---|---|---|
| CPU | GHz | W | cores | boards | chips | chips | cores | rates | SPECfp | % |
| AMD Athlon 64 3500+ | 2.20 | 47 | 1 | 16 | 16 | 1 | 1 | **8.5** | 136 | -7 |
| AMD Athlon 64 3800+ | 2.40 | 47 | 1 | 16 | 16 | 1 | 1 | 9.1 | 146 | 0 |
| AMD Athlon 64 X2 3800+ | 2.10 | 50 | 2 | 8 | 8 | 1 | 2 | 14.1 | 113 | -23 |
| AMD Athlon 64 X2 4200+ | 2.20 | 50 | 2 | 8 | 8 | 1 | 2 | **15.6** | 125 | -14 |
| AMD Athlon 64 X2 4800+ | 2.40 | 47 | 2 | 8 | 8 | 1 | 2 | **17.0** | 136 | -7 |
| AMD Athlon 64 X2 5000+ | 2.60 | 95 | 2 | 8 | 8 | 1 | 2 | **18.4** | 147 | 1 |
| AMD Athlon 64 X2 5400+ | 2.80 | 95 | 2 | 8 | 8 | 1 | 2 | 19.7 | 158 | 8 |
| AMD Athlon 64 FX62 | 2.80 | 125 | 2 | 8 | 8 | 1 | 2 | **19.8** | 158 | 9 |
| AMD Opteron 280 | 2.40 | 95 | 2 | 4 | 8 | 2 | 4 | **32.5** | 130 | -11 |
| AMD Opteron 285 | 2.60 | 95 | 2 | 4 | 8 | 2 | 4 | **36.6** | 146 | 0 |
| AMD Opteron 2212 | 2.40 | 95 | 2 | 4 | 8 | 2 | 4 | **31.2** | 125 | -14 |
| AMD Opteron 2212HE | 2.40 | 68 | 2 | 4 | 8 | 2 | 4 | **31.2** | 125 | -14 |
| AMD Opteron 2216 | 2.40 | 95 | 2 | 4 | 8 | 2 | 4 | **37.3** | 149 | 2 |
| AMD Opteron 2220 | 2.80 | 95 | 2 | 4 | 8 | 2 | 4 | **43.4** | 174 | 19 |
| Intel Core 2 Duo E6300 | 1.86 | 72 | 2 | 8 | 8 | 1 | 2 | **17.0** | 136 | -7 |
| Intel Core 2 Duo E6400 | 2.13 | 72 | 2 | 8 | 8 | 1 | 2 | **19.3** | 154 | 6 |
| Intel Core 2 Duo E6600 | 2.40 | 72 | 2 | 8 | 8 | 1 | 2 | **22.0** | 176 | 21 |
| Intel Core 2 Duo E6700 | 2.67 | 72 | 2 | 8 | 8 | 1 | 2 | **24.5** | 196 | 34 |
| Intel Core 2 Quad Q6600 | 2.40 | 105 | 4 | 4 | 4 | 1 | 4 | 43.2 | 173 | 18 |
| Intel Core 2 Quad QX6700 | 2.67 | 110 | 4 | 4 | 4 | 1 | 4 | **48.1** | 192 | 32 |
| Intel Xeon 5130 | 2.00 | 65 | 2 | 4 | 8 | 2 | 4 | **35.1** | 140 | -4 |
| Intel Xeon 5140 | 2.33 | 65 | 2 | 4 | 8 | 2 | 4 | **40.9** | 164 | 12 |
| Intel Xeon 5150 | 2.67 | 65 | 2 | 4 | 8 | 2 | 4 | **46.7** | 187 | 28 |
| Intel Xeon 5160 | 3.00 | 80 | 2 | 4 | 8 | 2 | 4 | **52.2** | 209 | 43 |
| Intel Xeon E5310 | 1.60 | 80 | 4 | 2 | 4 | 2 | 8 | **56.4** | 113 | -23 |
| Intel Xeon E5320 | 1.86 | 80 | 4 | 2 | 4 | 2 | 8 | **65.7** | 131 | -10 |
| Intel Xeon L5320 | 1.86 | 50 | 4 | 2 | 4 | 2 | 8 | **65.7** | 131 | -10 |
| Intel Xeon E5335 | 2.00 | 80 | 4 | 2 | 4 | 2 | 8 | **70.9** | 142 | -3 |
| Intel Xeon E5345 | 2.33 | 80 | 4 | 2 | 4 | 2 | 8 | **81.9** | 164 | 12 |

**Table 2.** In the nodes column gives the power consumption of a system (with 16 cores) for the computing nodes (sum of CPU, motherboard and memory). The power required for cooling and the total power are also given. The values in the Perf/W column are obtained by dividing the system performance values (see table 1) by the total power. A cost indication of a cluster is given in the cost column, which is based on retail prices found on the Internet (exclusive VAT and for Europe region). The values relative to a system equipped with AMD Athlon 64 3800+ processors are also given in percent.

| CPU | Power | | | | Cost (nodes only) | | Perf/W | Perf/€ |
|---|---|---|---|---|---|---|---|---|
| | Nodes | Cool | Total | Perf/W | € | Perf/€ | % | % |
| AMD Athlon 64 3500+ | 1320 | 100 | 1420 | 0.10 | 2752 | 0.050 | -7 | -4 |
| AMD Athlon 64 3800+ | 1320 | 100 | 1420 | 0.10 | 2826 | 0.052 | 0 | 0 |
| AMD Athlon 64 X2 3800+ | 757 | 100 | 857 | 0.13 | 1936 | 0.058 | 28 | 13 |
| AMD Athlon 64 X2 4200+ | 757 | 100 | 857 | 0.15 | 2256 | 0.055 | 42 | 7 |
| AMD Athlon 64 X2 4800+ | 730 | 100 | 830 | 0.16 | 2616 | 0.052 | 59 | 1 |
| AMD Athlon 64 X2 5000+ | 1153 | 100 | 1253 | 0.12 | 2696 | 0.055 | 14 | 6 |
| AMD Athlon 64 X2 5400+ | 1153 | 100 | 1253 | 0.13 | 3008 | 0.052 | 22 | 2 |
| AMD Athlon 64 FX62 | 1417 | 100 | 1517 | 0.10 | 5936 | 0.027 | 2 | -48 |
| AMD Opteron 280 | 1065 | 240 | 1305 | 0.10 | 5540 | 0.023 | -3 | -55 |
| AMD Opteron 285 | 1065 | 240 | 1305 | 0.11 | 6436 | 0.023 | 9 | -56 |
| AMD Opteron 2212 | 1065 | 240 | 1305 | 0.10 | 4960 | 0.025 | -7 | -51 |
| AMD Opteron 2212HE | 827 | 240 | 1067 | 0.12 | 8800 | 0.014 | 14 | -73 |
| AMD Opteron 2216 | 1065 | 240 | 1305 | 0.11 | 6064 | 0.025 | 11 | -52 |
| AMD Opteron 2220 | 1065 | 240 | 1305 | 0.13 | 8264 | 0.021 | 30 | -59 |
| Intel Core 2 Duo E6300 | 950 | 50 | 1000 | 0.14 | 2096 | 0.065 | 32 | 26 |
| Intel Core 2 Duo E6400 | 950 | 50 | 1000 | 0.15 | 2216 | 0.070 | 50 | 35 |
| Intel Core 2 Duo E6600 | 950 | 50 | 1000 | 0.18 | 2472 | 0.071 | 71 | 38 |
| Intel Core 2 Duo E6700 | 950 | 50 | 1000 | 0.20 | 2920 | 0.067 | 91 | 30 |
| Intel Core 2 Quad Q6600 | 691 | 50 | 741 | 0.23 | 2664 | 0.065 | 127 | 26 |
| Intel Core 2 Quad QX6700 | 713 | 50 | 763 | 0.25 | 4052 | 0.047 | 146 | -8 |
| Intel Xeon 5130 | 801 | 240 | 1041 | 0.13 | 5340 | 0.026 | 31 | -49 |
| Intel Xeon 5140 | 801 | 240 | 1041 | 0.16 | 6220 | 0.026 | 53 | -49 |
| Intel Xeon 5150 | 801 | 240 | 1041 | 0.18 | 7740 | 0.024 | 75 | -53 |
| Intel Xeon 5160 | 933 | 240 | 1173 | 0.18 | 8748 | 0.024 | 73 | -54 |
| Intel Xeon E5310 | 537 | 120 | 657 | 0.17 | 3918 | 0.029 | 67 | -44 |
| Intel Xeon E5320 | 537 | 120 | 657 | 0.20 | 4078 | 0.032 | 95 | -38 |
| Intel Xeon L5320 | 405 | 120 | 525 | 0.25 | 4238 | 0.031 | 144 | -40 |
| Intel Xeon E5335 | 537 | 120 | 657 | 0.22 | 4898 | 0.029 | 110 | -44 |
| Intel Xeon E5345 | 537 | 120 | 657 | 0.25 | 5406 | 0.030 | 143 | -41 |

The performance of the processors is based on the values provided by Standard Performance Evaluation Corporation (SPEC) [2]. The SPECfp_rate2000 performance benchmark is for comparing the floating point capabilities of computer while running multiple copies (one per CPU core) of a program. The benchmark consists of a suite of programs with different characteristics. The SPECfp_rate2000 result is the average value of the computer run times relative to a base system. For our special case of MCNP calculations, only one program of the suite was selected, i.e. the SIXTRACK program which is used for high energy nuclear physics accelerator design [3], because of the similarity with MCNP (physics modelling and use of random numbers).

Besides clock frequency, the performance depends on cpu architecture. Vendors are very careful in defining CPU prices for a specific type: prices are nearly proportional to performance for clock frequencies at the lower and middle range of a product range. Combined with the cost for motherboard and memory (minimum node configuration) an optimum Perf/€ is often observed close to the top of the range.

The new Intel core 2 is an example of a very effective design. MCNP calculations with an Intel Core 2 Duo E6400 test set-up (one dual core CPU) and the JRC Ispra cluster with 4 AMD Athlon

64 3800+ processors are close to tabulated performance values: the Intel processor is 29% faster, which is slightly less than the 35% value obtained with SPECfp_rate2000 for the SISTRACK code. One E6400 processor (released July 2006) is 29% faster than two single core A64 3800+ processors (released March 2005). When running two MCNP calculations simultaneously, one on each core, the measured power consumption of the test system (2 Gbyte RAM, Hard-disk and high efficiency power supply) was only 90W.

Although a cluster costs more than only the nodes (CPU, motherboard and memory), for instance there are the costs for assembly, cabinets and power supplies, the values in table 2 suggest that a cluster using processors for the consumer market (AMD Athlon and Intel Core 2) will provide more value for money than when using processors for the server market (AMD Opteron and Intel Xeon). Both motherboards and memory (registered) are more expensive for server processors. The extra costs are not completely compensated by the fact that server motherboards can be equipped with two processors. A fixed value of 1 Gbyte of memory per core has been used, providing enough memory for RAM disks and system memory.

Processors with more cores are more energy efficient, and those for the server market more than for the consumer market. Some of this energy efficiency is wasted by the air-cooling of 1U rack unit and blade servers. Especially the small fans or blowers with high rotating speeds used in 1U rack units are power hungry. In table 2 the values in the Cool column give an estimate of the power required by the fans of the server processor based systems. The systems with processors for the consumer market are cooled with liquid cooling, which compensates for the less efficient processors.

## 3. Liquid cooling

Besides being more efficient than cooling with air, liquid cooling provides a way to stack consumer market motherboards at a small pitch (45mm), extracting the heat produced by the processors. Large, low speed fans provide a much lower air-flow than that used in 100% air-cooled systems, which is used to extract the residual heat produced by the motherboard and memories. The heat containing liquid can be easily transported to large heat exchangers and water chillers, placed inside or outside the building.

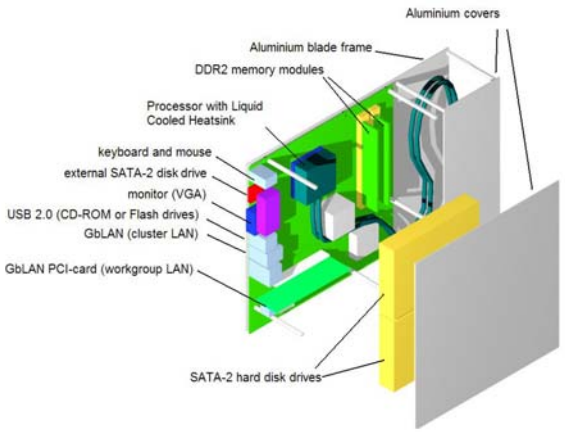


**Figure 1** Exploded view of a server node with two disk drives and a GbLAN PCI-card for connection to a workgroup LAN. The integrated GbLAN interface is used for connection to the disk-less nodes via a Gigabit switch (not shown).

Figures 1 and 2 show some aspects of a 19” rack mountable solution with space for maximum 9 diskless nodes. The height of the cabinet is only 5U. A pump unit (2U height) can serve 4 units. A standard rack may house 5 node units and 2 pump units: giving a total of 45 motherboards.

The mounting holes of the blade frames are according to the micro-ATX standard: the system can be re-used with the future generation of processors and motherboards.

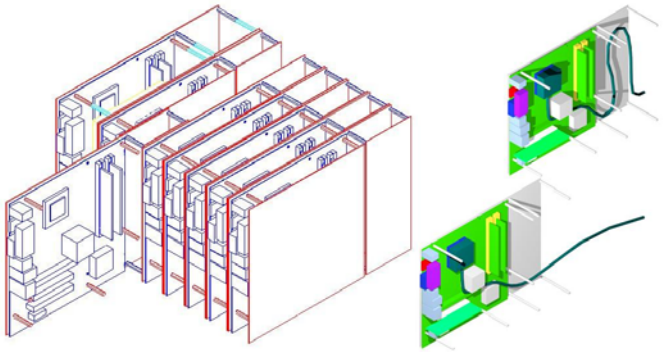

**Figure 2** On the left picture shows a diskless node in the maintenance position, granting access to the motherboard. The configuration shown is that of one server node (far left) and seven disk-less nodes. On the right the unique design of the liquid cooling supply and drain tubes.

## 4. Linux operating system

Stability, flexibility of kernel configuration and cost are well known positive aspects of the Linux operating system. Linux also provides a TFTP server, which is used for the remote boot of the diskless computer nodes, and rsh/ssh servers used by PVM to spawn tasks on the nodes. Moreover, special software packages (open source) are available for cluster management. The kernel has been recompiled to increase the RAM disks of the nodes to 500 Mbyte, which is enough for the operating system and MCNP-PTA pulse train files.

## 4. Diskless nodes

RAM disks are much faster and consume less power than disk drives. A 1 Gb DDR2-667 RAM module uses 1.9 W, which is one fifth of the power consumed by a hard disk (9.5 W). Moreover it is a 100% solid semiconductor device, without mechanical moving parts. Hard-disk in the contrary are subjected to wear. One of the conclusions in "Disk failures in the real world: What does an MTTF of 1,000,000 hours mean to you?" [4] is: "For drives less than five years old, field replacement rates were larger than what the datasheet MTTF suggested by a factor of 2–10. For five to eight year old drives, field replacement rates were a factor of 30 higher than what the datasheet MTTF suggested". The mean time to failure (MTTF) of disk drives, as specified in their datasheets, ranges from $10^6$ to $1.5x10^6$ hours, suggesting a nominal annual failure rate of at most 0.88%. In the field, annual disk replacement rates typically exceed 1%, with 2-4% common and up to 13% observed on some systems. In other words, there is a fair probability that one or more disks must be replaced in a period of 3 year for a cluster with 8 disk-based nodes.

## 5. Conclusions

With liquid cooling a compact cluster can be build using consumer market computer hardware. Diskless nodes and the liquid cooling compensate for the slighlty larger power dissipation of those processors with respect to the server market ones. The use of RAM disks increases the MTTF of a cluster. Processors with increasing number of cores result in a better energy effieciency and price/performance ratio. Form factor standards have a longer lifer than computer hardware, which is in continous evolution. The compact Beowulf cluster system allows to update part of the hardware (CPU, memory and motherboards) while keeping the infrastructure (cabinets and cooling).

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# 3D Scene Change Detection from Satellite Imagery

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***Abstract:***

*The use of reliable change detection from remote sensory data coming from Earth Observation (EO) technologies is one of the main challenges in security research. This paper describes the extraction of Digital Surface Models (DSM) from high resolution stereo imagery. It describes the algorithms used and the optimisations introduced to deal with man-made structures. The second part of the paper concentrates on the detection of changes from the different DSMs including the introduction of quality values for robust 3D change detection. Representative examples from real sites will be presented.*

**Keywords:** 3D, DSM, stereo, satellite imagery, earth observation, change detection

## 1. Introduction

One of the main open issues in using sensory data coming from Earth Observation (EO) technologies is the accuracy and timeliness of using automatic change detection methods in comparison to visual interpretation of changes, which is still the traditional method in several operational applications of remote sensing. While most of the changes are optically visible and an image analyst will be able to detect changes, automatic processes are difficult to achieve due to problems of registration and area/angle coverage of image series and the detected changes are often difficult to quantify.

One possibility to cope with the above mentioned problems is the use of 3D dimensional models of the facility. Stereometric data acquired by very high resolution optical sensors is a very promising source to extract 3D surface models. Nonetheless especially man-made structures cause problems in this approach resulting in significant variations using different algorithms or different parameter settings for one algorithm. This is mainly due to sudden height changes in the DSMs at the edges of buildings. These discontinuities result in smaller or larger occluded areas depending on the angle of acquisition. The development of algorithms which explicitly handle such occluded areas is an ongoing research.

## 2. Digital Surface Model Extraction

For the DSM extraction two Quickbird basic stereo pairs are employed in this study. A DSM is a height model which includes the elevation of the bare earth and the ”extra” elevations caused for example by vegetation (e.g. trees) and human activity (e.g. buildings or excavations).

The data is available in panchromatic and multispectral quality and has a resolution of up to 61 cm for the panchromatic stereo pair and up to 2.44 m for the multispectral one.

The first stereo pair was acquired from different orbits with a time difference of one month in summer 2004 while the other one was acquired along track in November 2005. The time difference of one month in the 2004 stereo pair causes additional problems in the 3D extraction process (e.g. by different light conditions, different appearance of agricultural areas, changes in the covered area, …). The extraction is performed at the JRC, Ispra - Italy and the Joanneum, Graz - Austria. JRC used the Leica Photogrammetry Suite (LPS), Joanneum Research the Remote Sensing Software Package Graz (RSG) for DSM extraction.

For DSM extraction using LPS, a block file has to be created. It contains information about the ellipsoid, the projection and the sensor type of the image data. The block file is a project file, which differs with the type of the imported data. Three different types of block files were possible to realise: “mixed sensor”, “orbital push broom” and “Quickbird RPC” .The project files “orbital push broom” and “mixed sensors” are based upon a camera model while the block file of the type “Quickbird RPC” is based upon a replacement camera model. RPC stands for rational polynomial coefficients and is a specific case of the Rational Function Model (RFM). The RFM represents the ground to image relationship of the physical sensor by third order polynomials but does not carry physical meanings of the imaging process. It allows a fast and quite precise rectification of satellite data even without accessible ground control points [1].

Due to its easy use and ground independency the “Quickbird RPC” block file occurs to be the most appropriate choice [2]. After the block file is created, the Quickbird stereo data is imported. In the next step the information on exterior and interior orientation of the image has to be included. In case of the employed “Quickbird RPC” this information is modelled by the RPC data. To refine the provided RPC data and increase the precision of the resulting location of points in the ground coordinate system a block triangulation / bundle adjustment has to be preformed. During the triangulation process the x-, y- and z coordinates of each tie point are calculated.

The next step is the DSM extraction. The configuration within the DSM extraction dialog allows to modify the output resolution of the DSM and to define specific scanning strategies for single parts of the scene (e.g. high mountainous, middle mountainous, flat, low urban . . .). The single strategies differ in the size of their search window size, their correlation window, their correlation coefficient limit, the strength of filtering and the assumed topography type and object type. [3]

In the given case the area is subdivided into a high mountainous, a middle mountainous, a flat, a rolling hill and a low urban part. For the stereo pair acquired in 11/2005 this predefined scanning parameter provides a fair result. For the stereo pair acquired in summer 2004 a modified scanning strategy is applied to suppress more errors. Nevertheless the resulting DSM for the 2004 data is poor. Figure 1 shows a screenshot of the “scanning strategy dialog” which was adjusted for the pseudo stereo pair from summer 2004. The visually best results for the part of the DSM covering the nuclear technology centre were reached for the stereo pair of 11/2005 using the panchromatic stereo pair and the predefined low urban scanning strategy for the extraction. For 2004 the best result were reached, combining the multispectral stereo pair DSM based on the low urban scanning strategy with the panchromatic stereo pair DSM based on the adjusted scanning strategy “special urban 4”.

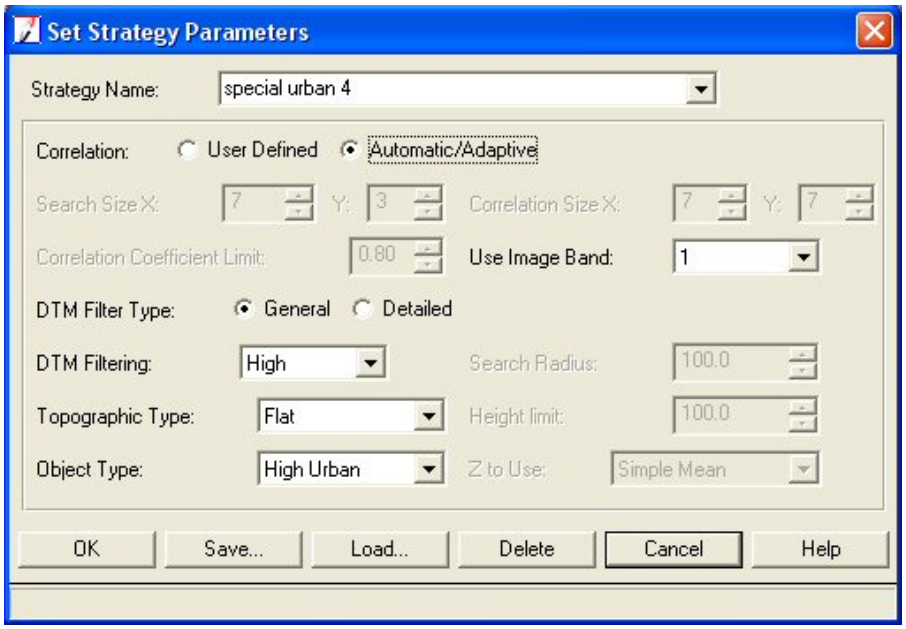


**Figure 1:** DSM Scanning Strategy "special urban 4"

The DSM resolution extractable with the LPS is limited mainly by two factors. The first factor is the resolution of the source data and the extraction algorithms the software uses. The second one seems to be the hardware requirements of the LPS. The increase of the spatial resolution for the whole scene causes a software crash. Apparently, the resolution can only be improved for extracts of the scene. In this way it is possible to extract DSMs covering an area of about 6 km² with a resolution of 2.5 m.

The DSMs extracted at the Joanneum using the RSG algorithms have a resolution of 1 m. Basically a similar procedure as using the LPS has to be followed. After defining the two stereo models, the imaging parameters (in case of Quickbird the RPCs) of each model are refined using tie-points. The key step in every stereometric processing chain is the extraction of corresponding features in both stereo partners. In RSG this procedure is based on a so-called hierarchical feature matching approach [4] which up-to-now has no explicit occlusion handling. The results of the matching are pixel displacements (disparities) and a quality criterion of the matching. Two minimize mismatches the matching procedure is applied twice: once from stereo partner A to B (forward) and a second time from B to A (backward). The difference between the starting (pixel) position and the for- and backward matched position is called backmatching distance. This distance is again a very sensitive measure of the matching quality. Finally the disparities are converted into 3d coordinates and interpolated into a regular grid (DSM).

Display details of the extracted DSMs are shown in Figure 2.

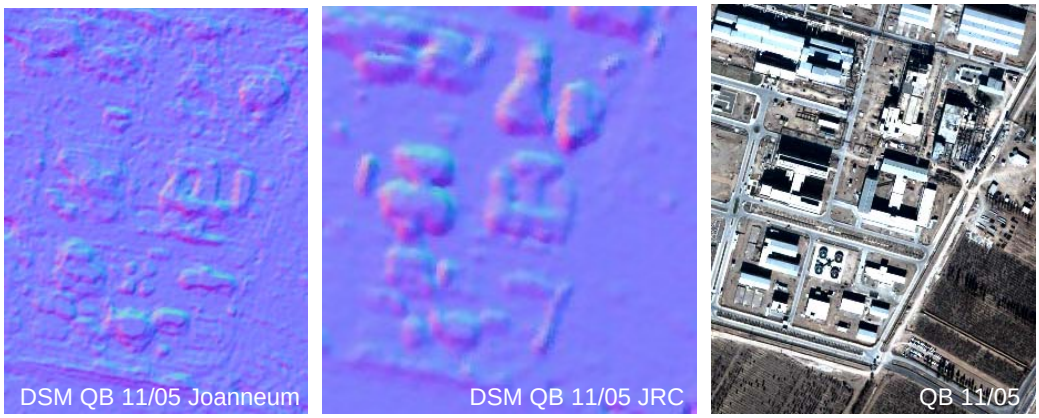


**Figure 2:** Display details of DSMs extracted from the Quickbird stereo pair of 11/2005

## 3. 3D Scene change detection

The 3D scene change detection based on the DSMs is preformed using the JRC Reconstructor software. Basically two different kinds of change detection were processed. Firstly the DSM extracted by the one method based on the imagery of 2004 and the one extracted with the same method based on the imagery of 2005 and secondly both DSMs of one year extracted with the two different methods are compared. The aim is to investigate whether it is possible to detect human induced changes in this way and how the results of the different extraction methods differ.

The DSMs are imported to the JRC Reconstructor as point clouds. First, pre processing filters are applied for hole filling, computation of normals and confidence values, edge detection and noise reduction.
Subsequently one point cloud is registered to the other. During the iterative registration process an optimised transformation is performed. Finally a transformation containing six parameters (three translation parameters and three rotation parameters) is suggested. Additionally the mean error of the suggested transformation and the error histogram is given. An example for the registration dialog of the JRC Reconstructor is shown in Figure 3.

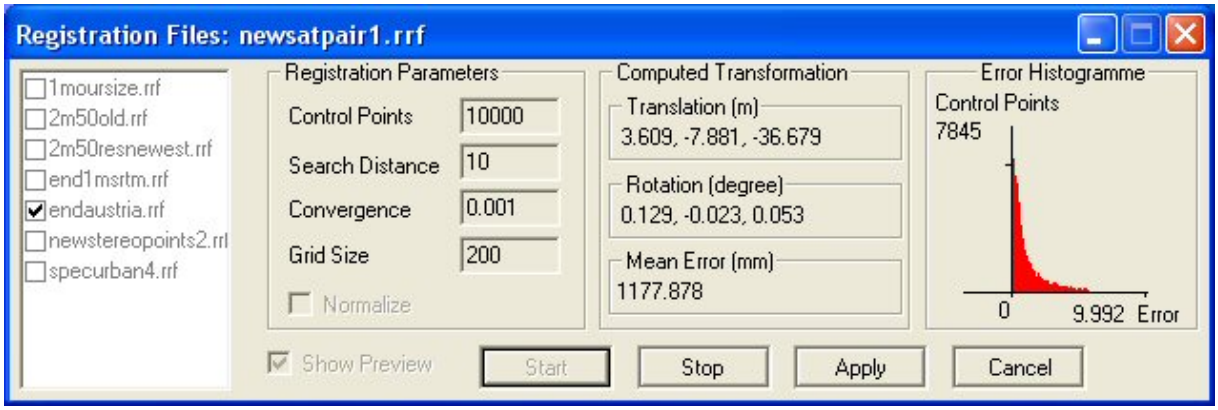


**Figure 3:** Registration dialog of the JRC Reconstructor

For the change detection process the reference point cloud has to be triangulated to a surface. Then the size and handling of the reference surface is optimized by applying an octree division algorithm on the created surface.

After that the change detection itself can be preformed. During the detection process the shortest distance between every point in the inspection point cloud and the reference surface is calculated and saved as a texture for the inspected point cloud. The single steps of the 3D change detection procedure are shown in Figure 4.

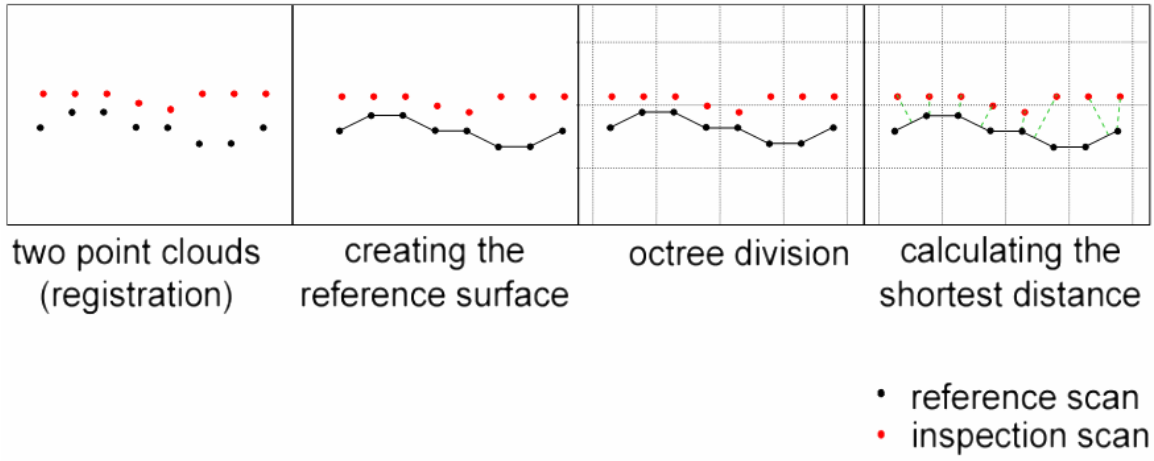


**Figure 4:** Workflow of the change detection procedure

An extract of the change detection results for the DSM based on the 2004 Quickbird pseudo stereo pair and the DSM based on the 2005 Quickbird stereo pair is shown in Figure 5.

## 4. Open issues with 3D scene change detection and DSM extraction

Both DSM extraction methods differ in their results, as to resolution, noise and quality. Based on the change detection results, the average absolute vertical difference between LPS and RSG is calculated to 2.1 m for the 2004 stereo pair and to 1.26 m for the 2005 stereo pair. The difference can be explained by the bad quality of the 2004 stereo pair and different ways the algorithms are affected by this.

The comparison of two change detection results of DSMs shows that e. g. a change of 10 m in the DSMs extracted at the JRC is only a change of 6 m in the DSMs extracted at the Joanneum. The height of the extracted building differs depending on the used method about 4 m.

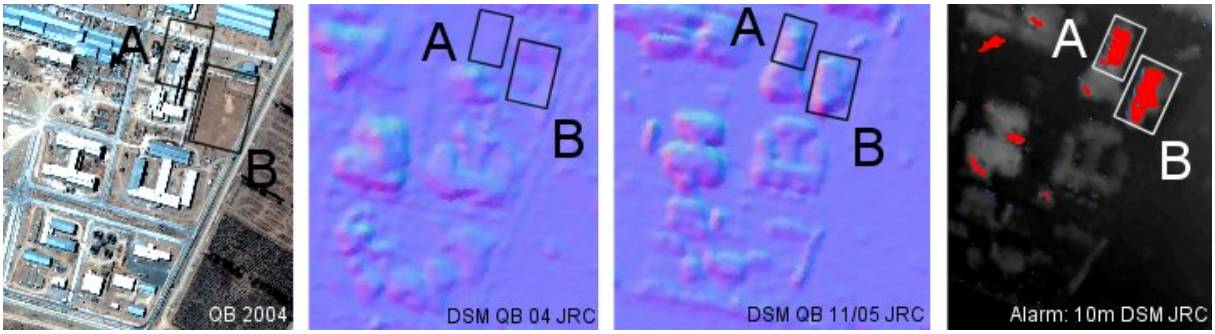


**Figure 5:** Detected changes

Another problem appears in both methods, as a building is not extracted in the DSMs based on the stereo pair from 2004. But the same building is extracted by the two methods using the 2005 stereo pairs. From this it follows that the software finds changes, which only exist as a difference between both DSMs but not in reality. An example for this effect is shown in Figure 5. Area A shows an area where a virtual change is detected while area B shows a real change or in other words: the building in area A existed already in 2004 but was not extracted while the building in area B did not exist in 2004 but in November 2005. The Quickbird image covering the same area in November 2005 is shown in Figure 2. This effect of a “virtual change” is probably caused by the bad quality of the DSM acquired in 2004 but nevertheless this “virtual change” is a source of error which has to be taken into account for the interpretation of DSM change detection results.

## 5. Conclusions and future work

The study shows that 3D scene change detection using very high resolution satellite stereo imagery is a feasible method to detect changes in elevation. But even if the results are promising it shows also the need for research in order to improve the results and reduce or compensate possible error sources. The heights of extracted buildings differ between the different DSM extraction algorithms and also the detection of “virtual changes” is an error source which cannot be excluded yet. However, this might be reduced by using only stereo pairs of good quality. Also the approach to introduce quality values like back matching to the DSM extraction process may lead to more reliable DSMs and result in a robust 3D scene change detection producing reliable results.

Current work is under way to evaluate the quality of the extraction methods and try to optimise it, using test areas with known, accurate and verifiable building geometry extracted from LIDAR data.

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# LIMES: Treaty Monitoring within the Security Dimension of GMES

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***Abstract:***

*GMES (Global Monitoring for Environment and Security) is a European initiative for the implementation of information services dealing with environment and security. It is based on the combination of Earth Observation (EO) data and in-situ measurements.*

*LIMES (Land/Sea Integrated Monitoring for European Security) is a GMES-funded Integrated Project proposing an exploratory methodology and implementation roadmap to organize the European EO capacities for security. LIMES started in December 2006 and includes a work package on Treaty Monitoring.*

*The objective of the Treaty Monitoring work package is to provide an integrated framework and platform supporting the verification of treaty compliance. This will be achieved by having a GIS-based platform capable of integrating, processing and analysing data and documents from multiple sources, including satellite imagery, site modeling, open source information, reports, etc. The Treaty Monitoring work package selected the Non-Proliferation Treaty (NPT) as a test case. A demonstration of the platform will be made using a relevant European site. The developments of this work package may be relevant to those involved in Treaty Monitoring activities, namely the European Commission (DG-TREN), the EU Council (SITCEN) or the IAEA.*

*The paper describes the objectives of LIMES and in particular of the Treaty Monitoring work package. It provides an overview of the functionalities and main benefits of the proposed platform.*

**Keywords:** GMES; LIMES; treaty monitoring; earth observation; GIS; data processing and integration

## 1. Background

GMES (Global Monitoring for Environment and Security) is a European initiative for the implementation of geo-spatial information services dealing with environment and security [1]. It supports decision-making by both institutional and private actors. Decisions could concern either new regulations to preserve our environment or urgent measures in case of emergencies and security threats. In order to take decisions, it is necessary to *anticipate*, *intervene* and *control*. GMES will integrate these functions by assembling the information received from Earth Observation (EO) satellites and ground based information in a reliable, valid and compatible manner and will make them available for user friendly exploitation. The services will be used by environmental agencies, local, regional, national and international authorities, civil protection organisations, etc. GMES is in its implementation phase and the objective is to gradually develop and validate a number of pilot operational services, based on selected R&D projects extending and strengthening the current actions.

LIMES (Land/Sea Integrated Monitoring for European Security) is a FP6 Integrated Project funded by the EU [2]. It aims at the development of pre-operational GMES services to support security management at EU and global level. LIMES services are clustered in three groups:

- Maritime surveillance including open, coastal water and sensitive cargo surveillance.
- Humanitarian relief and reconstruction includes services that cover the whole crisis cycle (disaster preparedness, operational support and support of reconstruction)
- Land and infrastructure surveillance includes land border monitoring, critical infrastructure surveillance, support to event planning and treaty monitoring. The services are based on the capacity of Very High Resolution satellites, used in conjunction with medium to high resolution data and aerial imagery, to enable critical 4D spatial analysis of updated reference data.

LIMES started in December 2006 and has a duration of 42 months. Prototype services will be implemented and evaluated after 18 months.

This paper describes the activities and objectives of the Treaty Monitoring work package. The work package is focusing on the Non-Proliferation Treaty (NPT) and aims to provide an integrated framework and platform supporting the verification of the NPT. It involves the following partners:

- European Commission Joint Research Centre (JRC), Italy
- Technische Universität Bergakademie Freiberg (TUBAF), Germany
- European Union Satellite Centre (EUSC), Spain

The Commissariat à L'Energie Atomique (CEA), France, contributes their expertise in SAR processing within the framework of a research collaboration.

## 2. Objectives and user benefits

The developments under the Treaties Monitoring work package are targeted at the image analyst in the field of Nuclear Safeguards who has the responsibility of collecting, managing and evaluating satellite imagery and extract the NPT relevant information. In this context, collateral data (e.g. Open Source information, internal databases, reports, GIS data) is becoming increasingly important to trigger, guide and support imagery-based analysis [3],[4]. The analyst generates a report on a location or country of interest and delivers it to the final user, which are high-rank decision makers at organizations involved in the monitoring of the NPT, e.g. IAEA, EU Council (SITCEN) or the European Commission (DG-TREN). It is the objective of the NPT monitoring platform to support the image analyst in the typical workflow for generating this report from multi-source and multi-temporal information (see Figure 1).

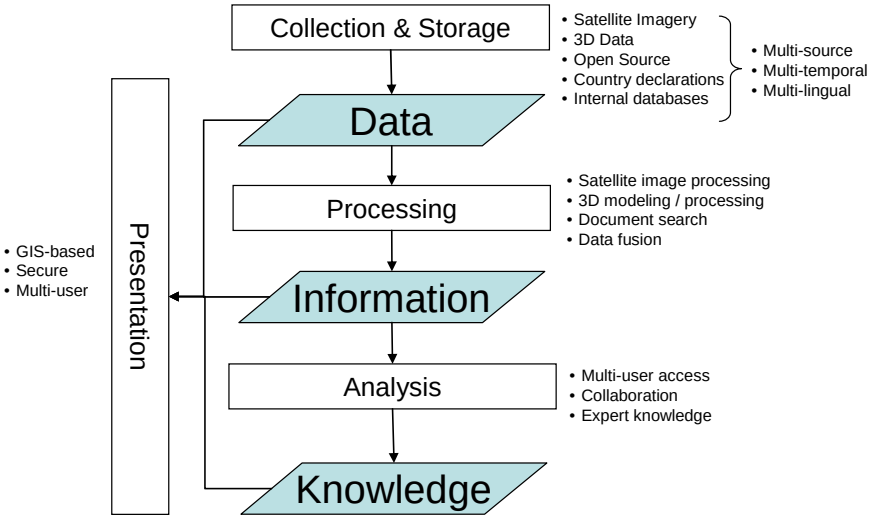


**Figure 1: Information flow in the context of NPT monitoring**

The first phase of the project aimed at establishing detailed user requirements. They were generated through discussions and previous experience with image analysts at IAEA, DG-TREN and EUSC. The outcome is summarized in the following paragraphs.

Since the advent of commercial high-resolution satellite sensors, remote sensing images provide essential information for recognizing and monitoring small-scale and short-term structural features of interest within nuclear facilities, for instance construction of buildings, plant expansion, changes of the operational status, preparation of underground activities etc [5]. Image data gathered by earth observation satellites has the advantage of being comparable, verifiable, taken remotely and continuously, which in turn is generally required for routine treaty monitoring applications. Commercial satellite imagery has therefore become indispensable in the verification process of the Non-Proliferation Treaty (NPT). Currently, the usage mainly relates to the implementation of the Additional Protocol (AP):

- monitoring of nuclear activities at known sites
  - identification of all buildings and facilities
  - understanding the purpose of each building
  - detection of any new constructions and changes
- detection of undeclared nuclear activities and facilities

However, current applications rely heavily on visual interpretation of satellite imagery with little use of automated processing. Furthermore, current analysis tools usually provide an isolated view on satellite imagery with poor integration of collateral data, such as Open Source information, GIS data, internal databases and reports.

Both, the number of nuclear sites monitored and the frequency of observations increase permanently. Moreover, the next generation of optical (e.g. QUICKBIRD- 3, IKONOS-3) and radar (e.g. TerraSAR-X, Radarsat-2) satellite sensors will come along with an enhanced spatial resolution. This increase in data requires higher effort regarding image (pre-) processing, analysis and interpretation. Computer-based techniques could be of great value in this regard. Though an automated system will not be able to replace an image analyst completely in the foreseeable future, she or he could benefit from (semi-) automation and transferability of digital image processing steps in order to detect and analyze significant features of interest much faster and more precisely. Therefore, the main improvements that will be addressed by LIMES are

- increased automation of image analysis (scene change detection, object classification, SAR processing);
- integration of multi-source and multi-temporal data (satellite imagery, Open-Source information, documentation, databases, GIS data).

# 3. Main features

## 3.1. GIS-based data integration

The NPT Monitoring platform will be a distributed, multi-user, GIS-based system that integrates existing databases, analysis and visualization tools (see Figure 2). The main characteristics are described as follows:

- A geo-database cross-links and integrates information from multiple sources, resolutions and time-frames, e.g. satellite imagery, 3D site models, open source information, reports, etc.
- A map-based user-interface provides a single point of access and allows the user to directly retrieve (non-spatial) information related to geographic features on the map.
- The geospatial information is served via the http protocol using standard formats in order to ensure compatibility with a maximum number of client applications.
- The platform enforces existing security constraints and ensures that each user can access only data and functionality for which he is authorized.
- The platform includes improved tools for automated, object-based processing of high-resolution satellite imagery and for efficient management, evaluation and correlation of Open Source information (see below for details).

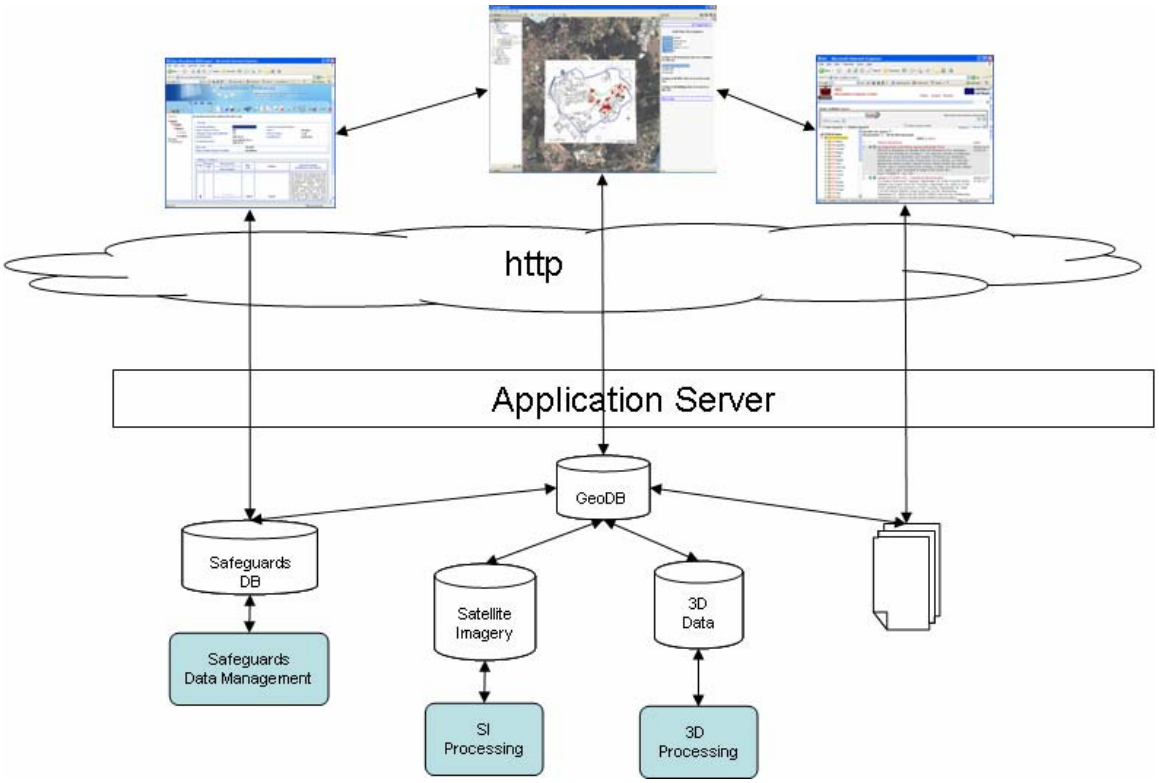


**Figure 2: System architecture**

## 3.2. Change detection

The platform includes a module for detecting nuclear safeguards relevant changes through multi-temporal analysis of medium- and high-resolution optical satellite imagery [6]. It covers the following processing steps:

- Geometric and atmospheric correction is essential for an exact pixel-by-pixel or object-by-object comparison during the change detection process. By means of geometric correction

algorithms, the image data can be registered to each other (image-to-image registration) or to a given map projection (geo-referencing).

- Wide-area scanning using medium resolution imagery is intended for the detection of potential nuclear-related undeclared activities and the detection of major changes within declared nuclear sites and their surrounding areas.
- Detailed change detection and analysis using high-resolution imagery for the monitoring of known sites. It combines pixel-based change detection and object-based image classification to detect changes in the areas of interest (see Figure 3). Emphasis is put on automated creation of standardized and transferable classification models.

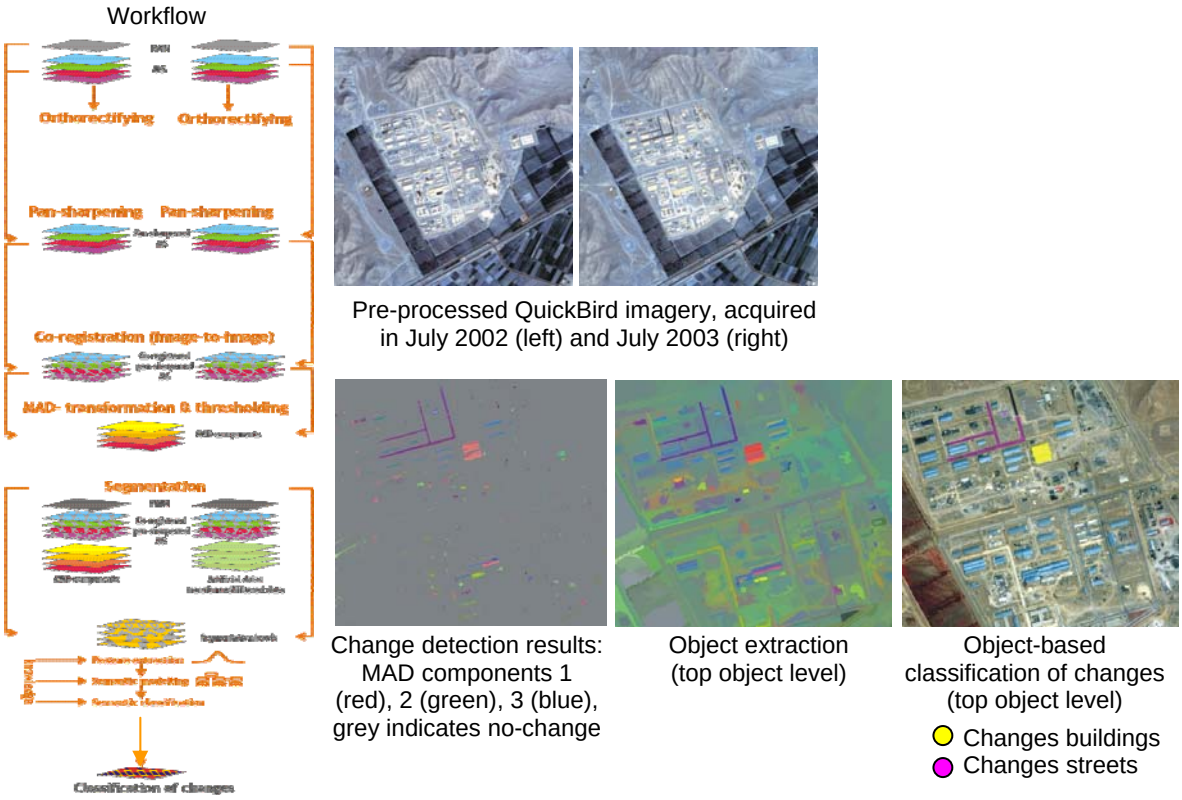


**Figure 3: Combination of change and thematic information for object-based change analysis (2002-2003 for parts of the Esfahan Nuclear Fuel Research and Production Centre).**

## 3.3. SAR imagery processing

SAR (Synthetic Aperture Radar) interferometry exploits the phase difference between two SAR images taken from different viewpoints and allows the generation of digital elevation models (up 10m accuracy with current data), displacement maps (centimetre accuracy) and coherence maps. SAR provides high quality images in all weather and time conditions.

In the near future several SAR satellites will provide X-band images with metric resolution. Due to their complex nature, SAR images are not suitable for visual interpretation by the human eye. Consequently, automated processes for detecting anomalies are required. The NPT monitoring platform will incorporate a software package which detects anomalies by analysing series of interferograms taken at different instants in time [7]. It compares the coherence map calculated from the current data set with the coherence predicted from previous measurements (see Figure 4).

Tests made with high-resolution SAR imagery from airborne sensors show that the techniques can be very useful for NPT Monitoring as soon as the high-resolution, spaceborne SAR imagery becomes available.

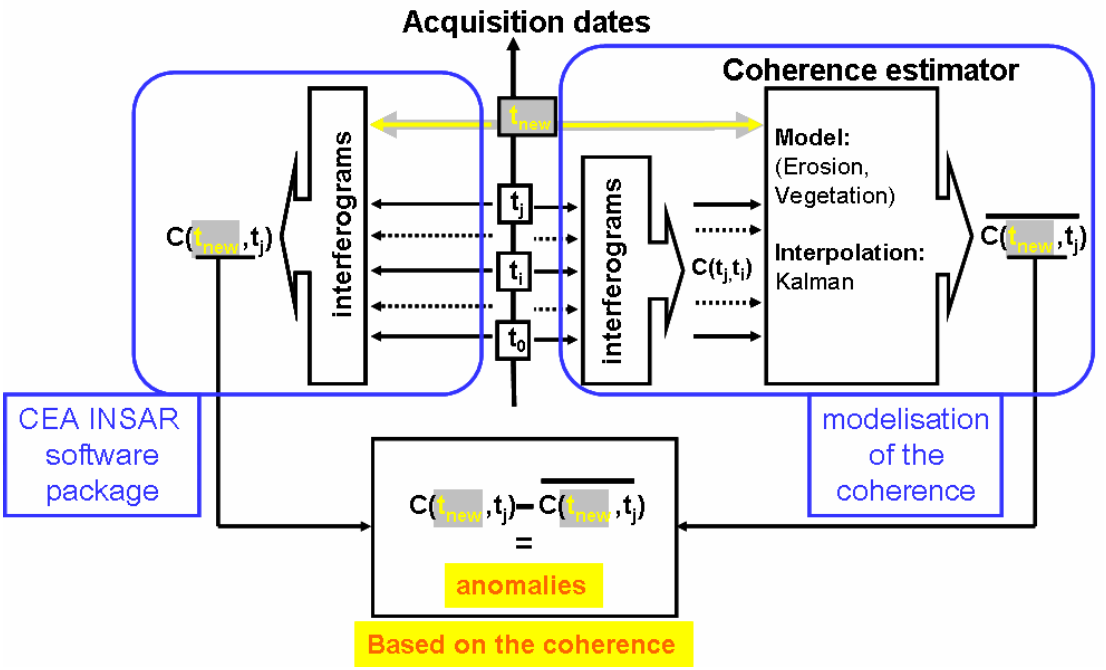


**Figure 4: Principle of anomaly detection from coherence image**

## 4. Demonstration case

The platform will be demonstrated and evaluated after the implementation of the prototype. A European nuclear site will be selected and relevant data of the site will be acquired (satellite imagery and collateral data). The platform will be installed at EUSC and used during the generation of a report regarding the selected test site. The image analysts at EUSC will evaluate the impact of the platform with respect to their work. IAEA will continue to provide advice and feedback for the evaluation and optimization of the platform.

## 5. Discussion

Commercial high-resolution satellite imagery is an important tool for monitoring the Non-Proliferation Treaty and, in particular, for the implementation of the Additional Protocol. The importance will further grow with improved spatial resolution of future optical and radar satellite sensors.

Currently, the image analyst relies primarily on visual interpretation of high-resolution optical imagery. The increasing amount of data, the need to detect undeclared activities in wide areas and new sensor types (e.g. high-resolution SAR imagery) require automated processing of satellite imagery in the context of Nuclear Safeguards. Furthermore, the tools should not only support the analysis of satellite imagery, but should allow the analyst to take a multi-source approach considering data from satellite sensors, Open Source information, in-situ measurements, internal databases and reports, etc.

The treaty monitoring activities of the LIMES project addresses these challenges through the implementation of a GIS-based platform capable of integrating, processing and analyzing data and documents from multiple sources. The platform will support the image analyst by providing automated change detection in optical imagery and improved information extraction from high-resolution SAR data.

## 6. Acknowledgements

LIMES is a FP6 integrated project funded by the European Commission under FP6-2005-SPACE-1/GMES SECURITY.

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# Safeguards manual: Nuclear materials handbook

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## National safeguards system

National safeguards system is compilation of network taking care of nuclear non-proliferation. To work effectively, all parts of this system must have clear idea what nuclear non-proliferation and nuclear materials safeguards means as well as know what their obligations are.

## Implementing safeguards at facility level

According to the Commission Regulation, any persons or undertaking shall maintain a system of accountancy and control of nuclear materials. To establish this system both international and national requirements shall be taking into account. Certain procedures, operational principles and instructions how these requirements can be met shall be described in nuclear materials handbook. In other words, NM handbook is a safeguards manual how the operator implements its safeguards at the facility level. Thus handbook is an essential part of operator's quality system. NM handbook is also useful for authorities, who can verify the effectiveness and functionality of operator's safeguards system by comparing the handbook and reality during normal inspections as well as in special safeguards system audits.

## Finnish approach

In Finland, STUK has regulated that the license holder is obliged to prepare nuclear materials handbook for material balance area or other accounting unit such as international transfers, and keep it up-to-date. License holder must include into the handbook instructions and guidelines how the nuclear materials accountancy, reporting and other control obligations are implemented in its facility. The handbook and essential changes to it must be submitted to STUK for approval. License holder is also obliged to organise independent audit to verify the correctness and currentness of the handbook at least every fourth year. Bigger operators such as NPPs and research reactor have prepared their own handbooks. For smaller operator having e.g. depleted uranium in shielding, STUK has prepared model handbook to meet the requirements.

## NM handbooks in Finland

- Fortum/Loviisa NPP
- Fortum/International transfers of NM
- TVO/Olkiluoto NPP
- TVO/International transfers of NM
- VTT FiR1 -research reactor
- Model handbook for small NM holders
- Posiva: Nuclear non-proliferation control in underground characterisation facility ONKALO

Keywords: NMAC, NM handbook, SG implementation, quality system, system audit

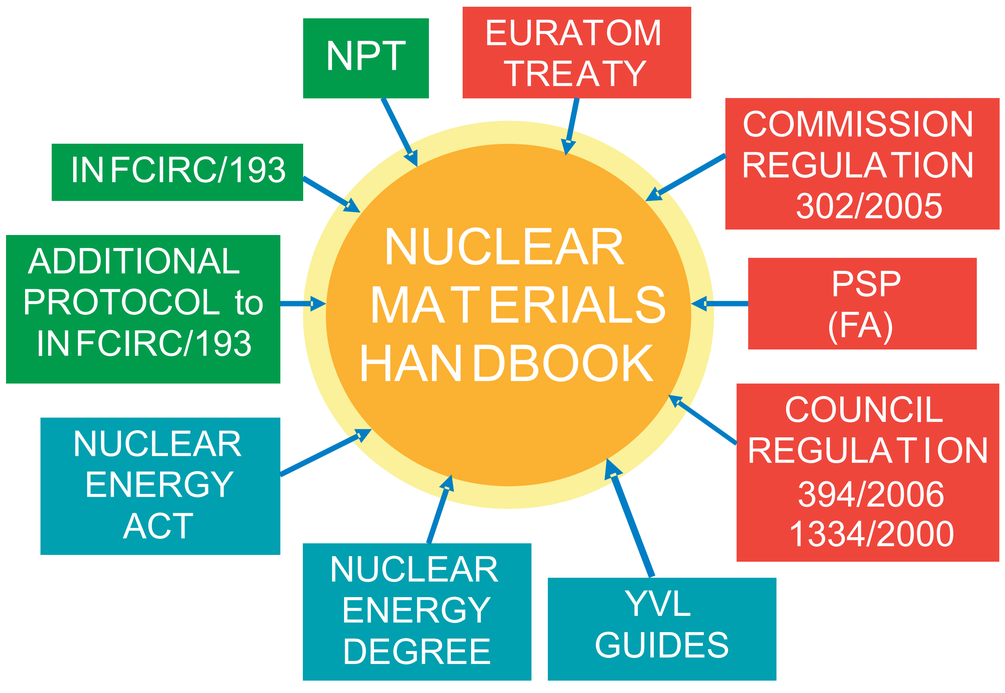


Fig. 1: International and national requirements concerning safeguards manual.

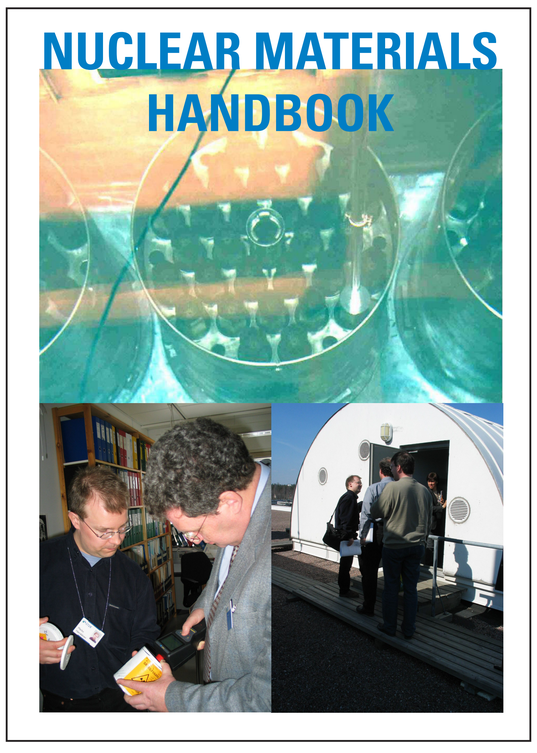


Fig. 2: NM handbook (model)

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2. Basis of safeguards
3. Roles and responsibilities
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# Uranium isotopic analysis of particles by Laser Ablation and Inductively Coupled Plasma Mass Spectrometry

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**Abstract:**

Particles analysis on swipe materials is a powerful tool regarding safeguard issues. Two techniques are currently used: the fission track and thermal ionization mass spectrometry (FT-TIMS) and the Secondary Ion Mass Spectrometry (SIMS). Tests have been performed in order to evaluate the potential of Laser Ablation and Inductively Coupled Plasma Mass Spectrometry (LA-ICPMS) technique for uranium isotopic analysis of particles. For this study, we have use an Alfamet femtosecond laser (Novalase, France) with a wavelength of 1030 nm. As this laser allows a large range of repetition rate (from 1 to 10,000 Hz), among others, we have investigated the effect of this parameter on the degradation of the matrix around particles, but also on the ion beam intensity, and on the signal length. The ICP-MS used in this study is a Thermo "X-series" quadrupole-based ICP-MS equipped with a single detector. As the signal is transient and noisy, the precision of isotopic ratios is largely dependant on the integration time used during the measurement of each isotope. Ideally this could be overcome by using multi-collector instrument in the next stage of this work. The results we will present show that this first attempt is very promising, but there are some issues that need to be investigated more carefully such as the mass fractionation that occurs during the ablation, and the ionization in the plasma.

This technique allows a discrete sampling of a few micrometers as the SIMS, but the preparation of the sample is less restrictive than for SIMS technique. In comparison with classical FT-TIMS process, this approach is less time consuming, as there is no need to irradiate samples and to samples particles on the track etch detector.

**Keywords:** Laser ablation; ICPMS; uranium isotopes; particles

**Poster contribution**

# Safeguards Approach for the Vitrification of HLLW at the dismantled Karlsruhe Pilot Reprocessing Plant (WAK)

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***Abstract:***

*In the past international nuclear safeguards was focused on operating facilities and in these facilities namely on the product streams. The amount of nuclear material in waste was regularly measured or estimated by the plant operator but normally accepted by the IAEA without verification, if specified limits were respected. With the Additional Protocol (AP) in force, safeguards measures are now also applied on (in a technical sense) decommissioned plants and the conditioning process of nuclear material in waste will be safeguarded by the IAEA to a higher degree than in the past. This paper will describe in detail the situation at the Wiederaufarbeitungsanlage Karlsruhe (WAK).*

*Concerning the safeguards relevance of the process and the glass product, the IAEA had to accept that a credible diversion path does not exist for the nuclear material. The vitrification process itself has no possibilities of recovering U and Pu and, furthermore, on the sites of FZK and WAK reprocessing capabilities do not exist anymore. Keeping this in mind, a safeguards approach has been agreed in 1999. The safeguards measures have been discussed in detail within the last year between IAEA, EURATOM, the German Federal Ministry of Economics and Technology and WAK. As a result, EURATOM and IAEA will manage safeguarding VEK during the 1.5 years period of facility operation without any permanently installed equipment, and both inspectorates will be able to spend their limited resources on other safeguards challenges.*

**Keywords:** vitrification; HLLW; safeguards;

## 1. History

In 1956 the (now) Forschungszentrum Karlsruhe (FZK) was founded in the north of Karlsruhe, Germany. In the beginning the aim of this nuclear research centre was the development of nuclear reactors. In relation to this work also research on reprocessing and waste handling was established. In 1967 the construction of the Wiederaufarbeitungsanlage Karlsruhe (WAK) pilot reprocessing plant started. The aim of this facility was the testing of flow sheet variations and process components developed by institutes of the FZK. Furthermore, also staff training for the planned industrial scale reprocessing plant was necessary. WAK started its hot operation in 1971. During 31 campaigns 207 Mg of uranium and 1.16 Mg of plutonium originating from different German reactors were reprocessed. The average burn-up was approx. 17 GWd/Mg U, the peak value 40 GWd/Mg U. In 1989, the German utilities decided to stop the construction of the industrial scale Wackersdorf reprocessing plant. The result was the final shut down of WAK at the end of 1990. Fig. 1 shows the actual buildings on the WAK site. Furthermore, all the research and development activities concerning reprocessing were stopped at FZK. Today, only waste conditioning and intermediate storage facilities like the Hauptabteilung Dekontaminationsbetriebe (HDB) and the Institut für Nukleare Entsorgung (INE) are in operation. These facilities are necessary for the dismantling of the FZK owned research reactors as well as for WAK.

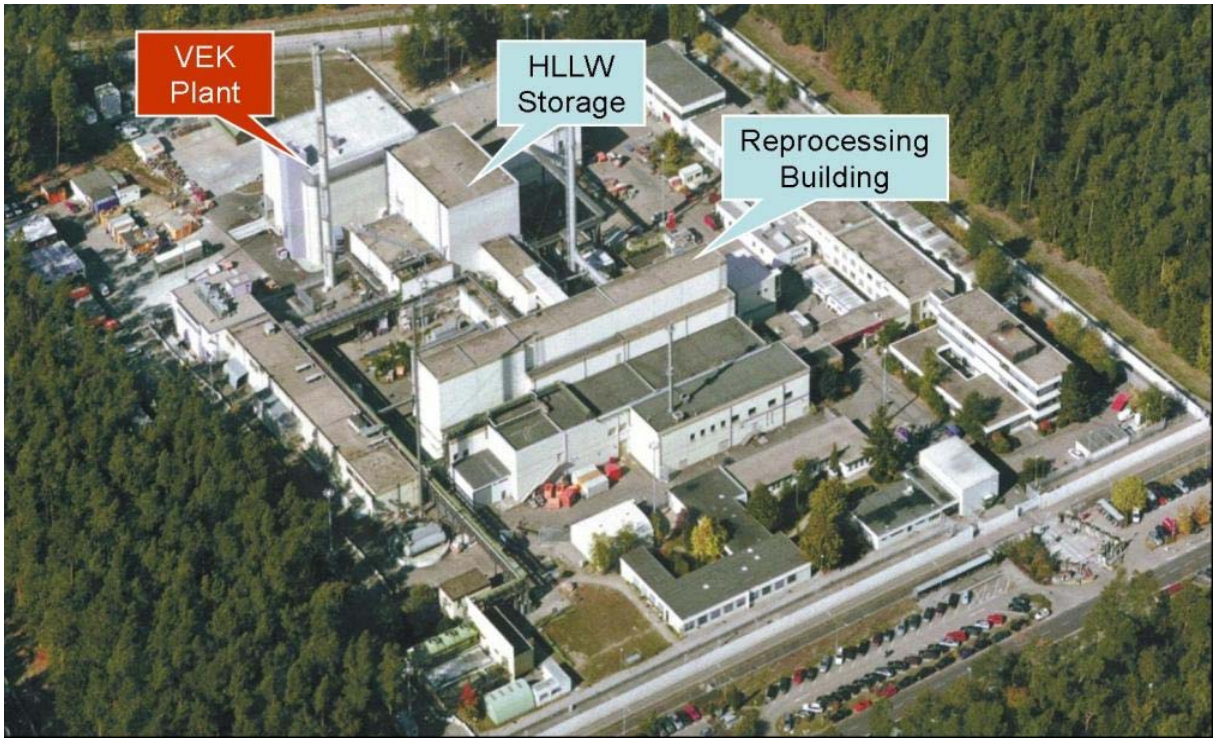


**Figure 1:** Aerial view on WAK site

**Figure 2:** Remote dismantling of the HLLW evaporator cell

**Figure 3:** HLLW evaporator cell after removal of the pipe penetration blocks

## 2. Status of the WAK Dismantling Project

After the final shut down of WAK at the end of 1990 the plant was rinsed and all separated plutonium and uranium were shipped off site. Starting in 1996, the equipment in the process building has been totally dismantled. First 12 systems only having a low activity level could be dismantled manually. In a second step the content of all process cells was dismantled by remote handling (Fig. 2) and also the necessary control systems were removed. Today even most of the pipe penetration blocks between the hot cells are already cut out (Fig. 3). In a first campaign hot spots were removed by abrasive methods. All liquid and solid wastes produced during the dismantling activities were shipped to HDB. The progress in dismantling was regularly verified during the inspections of EURATOM and the IAEA. An overview of the WAK dismantling project has been given in [1], remote dismantling is described in detail in [2].

Only one part of WAK is still in hot operation: In a separate building (LAVA) approximately 60 m³ of high level liquid waste (HLLW) with a total radioactivity of nearly 8E17 Bq are stored as “retained waste” to be conditioned on site. This vitrification project also has been established in 1996 and a new building for the Verglasungseinrichtung Karlsruhe (VEK) is finished and equipped, now waiting for the license of hot operation in 2007.

## 3. Design of VEK

### 3.1. Structure of the building

The vitrification of the HLLW is an essential step for the total dismantling and demolition of WAK. The VEK facility is only planned, constructed and licensed for this specific task that should be terminated within 1.5 years of operation. The equipment of the main process is installed in several hot cells as indicated in Fig. 4, which shows a longitudinal cross section of the VEK building. The HLLW receipt cell (1) contains two receipt tanks as well as the secondary liquid waste treatment. In the melter cell (2) the HLLW feeding vessel, the melter and the first two off-gas components (dust scrubber and condenser) can be found. The two off-gas treatment cells (wet/dry) are located behind the rear wall of the hot cells and therefore are not visible in Fig. 4.

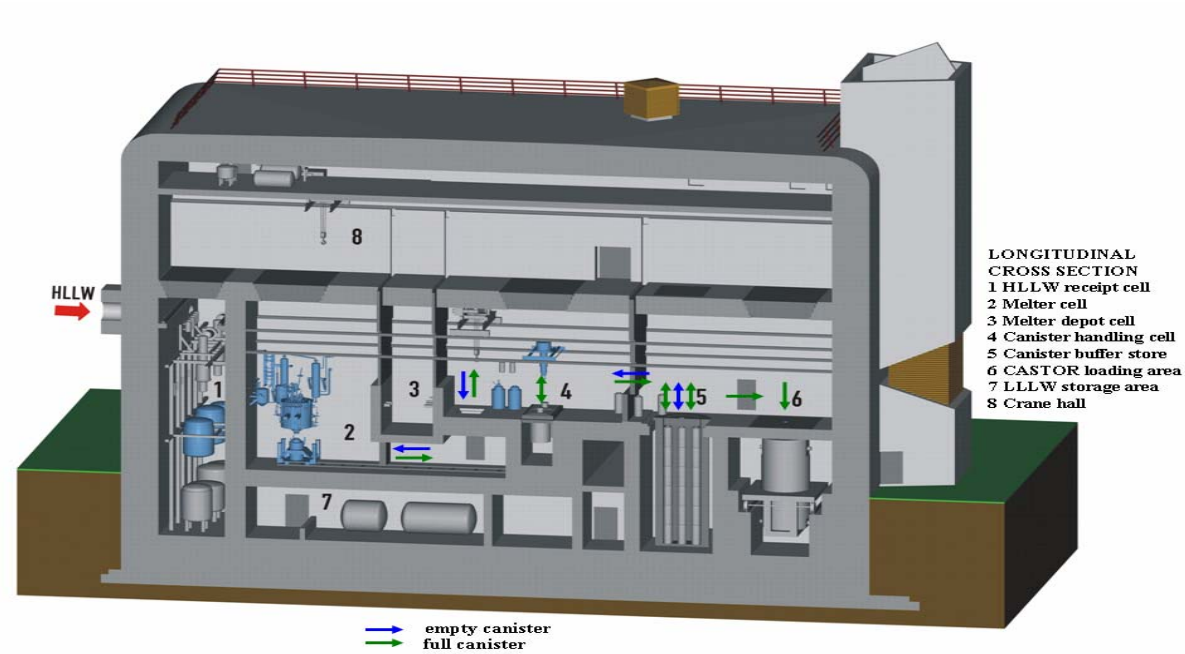


**Figure 4:** Longitudinal vertical cross section through the central area of the VEK building including the movement of empty and full glass canisters

In the unlikely event of a severe melter failure during hot operation, the old melter can be removed remotely and stored in the melter depot cell (3). A second melter was built and is ready for this replacement. In the canister handling cell (4) we find the cooling station, the automatic welding device for the canister lid and the decontamination unit. At the right of Fig. 4 the canister buffer store (5) and the CASTOR loading area (6) can be seen. The LLLW storage area (7) in the basement is used for liquids coming from the off-gas treatment. Heavy components in the cells may be replaced for maintenance reasons by using the equipment of the crane hall (8). The design of VEK has been already described in detail [3, 4].

### 3.2. Vitrification Process and Canister Handling

Looking at the flow sheet of vitrification, it is a straight forward process (Fig. 5): In LAVA a volume of 1.6 m³ HLLW is analyzed and transferred to VEK. In VEK approx. 40 l of ILLW (coming from the wet off-gas treatment) is added and the mixture is analyzed once more to verify the oxide content. The solution is transferred automatically in a small dosage vessel and than poured continuously on the surface of the melted glass. The raw glass itself is dosed batch wise as small pearls of glass frit. The HLLW is dried, calcinated and its chemical elements are incorporated as oxides into the glass matrix.

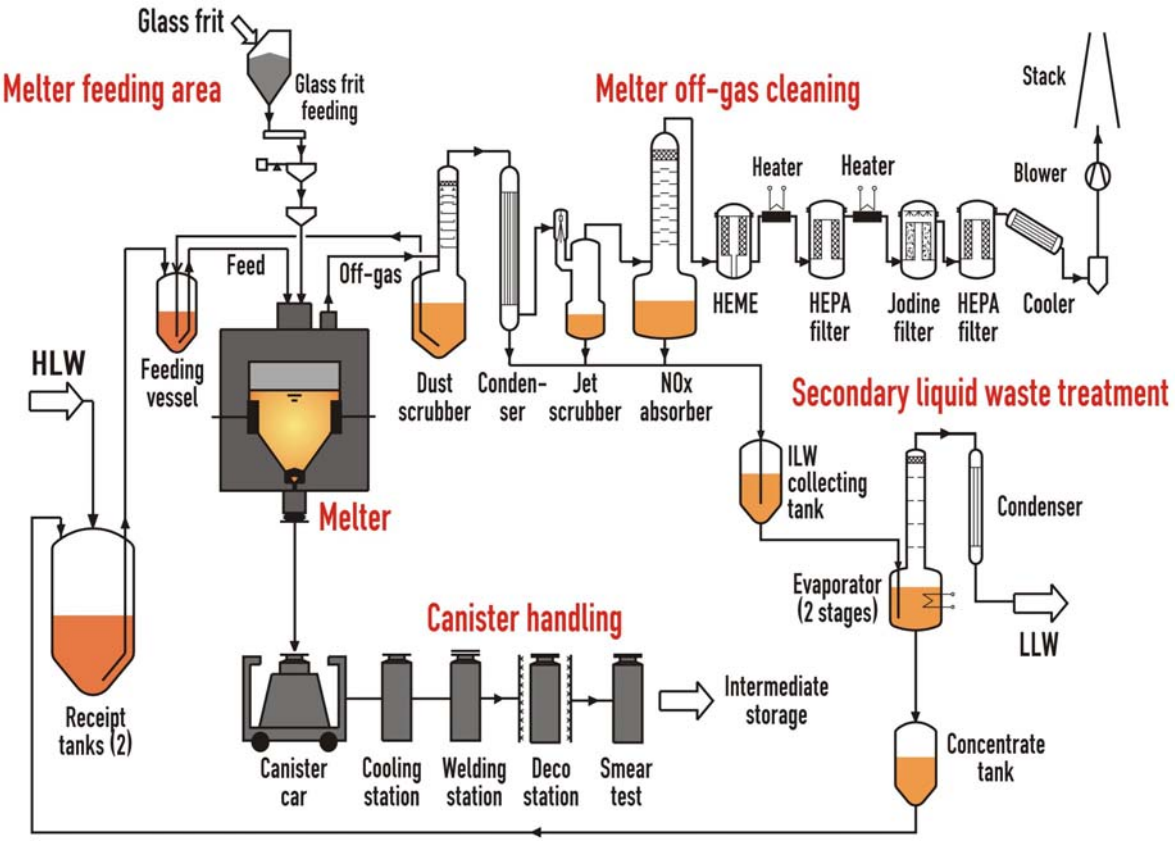


**Figure 5:** Simplified flow sheet of the vitrification process

The melter will contain about 400 kg of glass which will be filled every 15 hours (in four steps of 100 kg) into stainless steel canisters. The canisters are numbered and they will be weighed prior, during and after filling. The canisters will be transferred to the canister handling cell and there cooled down for several days. Then a lid is welded remotely on the canister and the canister is decontaminated. Prior to buffer storage neutron as well as beta and gamma dose rates are measured in the next cell.

The buffer storage consists of seven pipes each storing up to six canisters. The centre pipe will only be used for empty canisters so that 36 full canisters can be stored. Each 28 canisters will be loaded to a CASTOR cask. In total 130 canisters will be produced for which five CASTOR casks will be prepared and transported to an intermediate storage facility. In Fig. 4 the way of the full canisters is shown by green arrows, the way of the empty canisters by blue arrows. There are only three differences in

handling: empty canisters will not arrive in a CASTOR cask, they will not be measured for activity prior to buffer storage and they will not be treated in the canister handling cell.

## 4. Safeguards Approach

Concerning the safeguards relevance of the process and the glass product, it is obvious that a credible diversion path does not exist for the nuclear material. The vitrification process itself has no possibilities of recovering U and Pu and, furthermore, on the sites of FZK and WAK reprocessing capabilities do not exist anymore. The nuclear material content of the HLLW is well known to the inspectorates and may be verified once more prior to vitrification. As a consequence, a safeguards approach has been agreed in 1999. It has included the design verification of VEK, sealing of back transfer lines from VEK to LAVA (which are necessary for safety reasons), the authentication of the operator measurements on the product canisters and a quarterly inspection. These measures had been recognized as sufficient for a statement of compliance with the safeguards requirements.

Keeping the IAEA informed on the VEK project by the WAK annual activity program, final discussions between the inspectorates, the German Federal Ministry of Economy and Technology, and WAK on the details of the agreed measures started in early 2006 [5]. As a result, the following safeguards measures have been agreed:

1) Verification of the volume calibration in one of two VEK HLLW input tanks (already done)
2) Verification of U and Pu contents by sample taking and analysis in one of two LAVA HLLW storage tanks
3) Sealing of back transfer lines from VEK to LAVA
4) Physical inventory taking (PIT) prior to start of vitrification
5) Monthly inspection during hot operation of VEK
6) Taking additional five VEK input samples of liquid HLLW
7) Information about all relevant production data of each glass canister
8) Independent neutron measurements at the CASTOR casks with the filled glass canisters
9) Sealing of the CASTOR casks with the filled glass canisters during storage at WAK (prior to transportation to the intermediate storage facility)
10) PIT after termination of vitrification (possibly also an intermediate PIT will be necessary if the vitrification campaign will last more than one year)

Combining these measures, recognizing the missing reprocessing capabilities at the WAK and FZK sites and considering that the Additional Protocol is in force in the European Union, we do believe that the IAEA has sufficient qualitative and quantitative information to verify the absence of clandestine reprocessing in Germany.

## 4. Conclusion

The vitrification of the HLLW resulting from former nuclear fuel reprocessing is an essential step for the complete dismantling and demolition of WAK. The vitrification process has no capability of recovering fissile material. Furthermore, all research and development activities related to reprocessing have been definitely stopped at the FZK site. As a result, a credible diversion path for the fissile material in the HLLW no longer exists. With the Additional Protocol in force the IAEA gets sufficient information to verify the compliance with the safeguards criteria. As a consequence, EURATOM and the IAEA would be able to spend their limited resources on other safeguards challenges.

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# Estimation of Radiation Effects following a Radiological Dispersion Event

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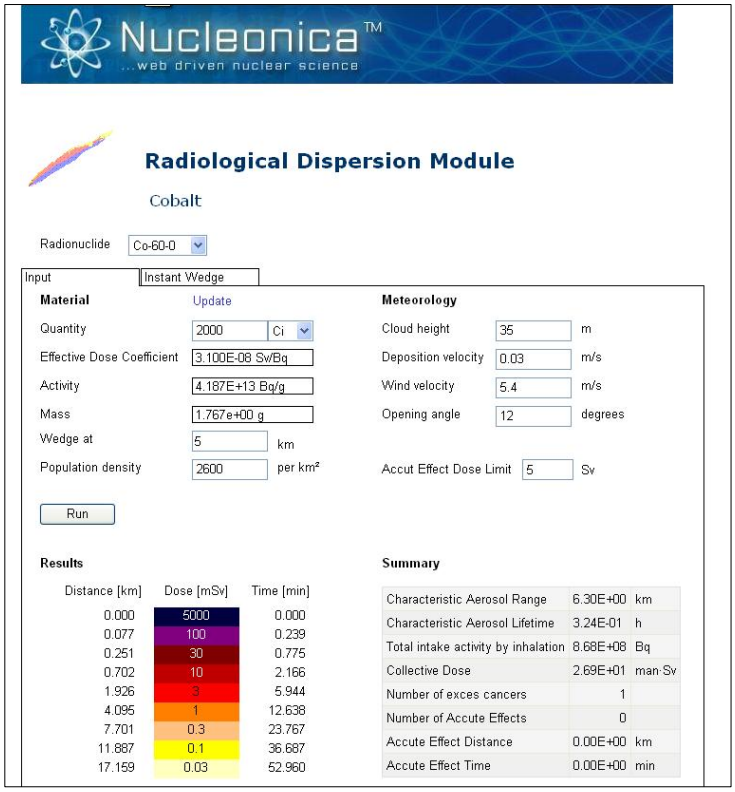
A software module has been developed within ITU's own web-based Nucleonica framework [1] in order to provide scientists and policymakers with quick and reliable estimates of radiation effects following a Radiological Dispersion Event (RDE).




**Modeling Approach (Wedge Model)**

Accounting for RDE phenomena involves two separate modeling stages:

1. Dispersion of radioactive aerosol under given meteorological conditions.
2. Public health consequences due to exposure to radioactive aerosol.

In order to describe the evolution of a radioactive aerosol plume resulting from an RDE, we utilise the wedge model [2], which gives a simple, intuitive account of the dynamics governing the dispersal of a radioactive aerosol cloud under given meteorological conditions. In this model, the plume evolution is described by a simple convolution of geometrical and aerosol deposition mechanisms.

In the second stage, we use effective dose coefficients (ICRP72) to relate total intake activity to full body dose over a fifty year time period. It is possible within this framework to make separate estimates for acute and chronic radiation effects.

**Nucleonica Module**

A full online implementation of the process described above has recently been completed and can be seen on the right-hand side of the poster. The top figure shows a screenshot of the user-friendly interface, including both radiological and meteorological input parameters and an output summary. A more detailed output can be seen in the bottom figure, which shows a dose profile comparison for a particular RDE scenario: The profile on the left corresponds to the Nucleonica module, whilst the one on the right is a comparison with the Lagrangian Particle code LASAIR [3].

**References**

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# AN OVERVIEW OF WASTE NUCLEAR MEASUREMENTS IN THE ATOMIC ENERGY COMMISSION FACILITY OF VALDUC

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ESARDA 29th Annual Meeting, Symposium on Safeguards and Nuclear Material Management, Aix-en-Provence (France), May 22-24, 2007

***Abstract***

The Valduc Center, which belongs to the CEA's Military Applications Division, generates solid wastes contaminated with alpha emitters.
In order to support and perform all the waste measurements, a central laboratory is designed to give a technical support that includes training, supervising and waste producers in-situ maintenance.
This Analysis and Waste Service's laboratory assumes the full responsibility of all wastes center's measurements, leaning on the primary producers.
A strategy has been implemented to provide a high degree of quality insurance: harmonized techniques and softwares, first measurement confirmed as soon as possible by a second one, laboratories inter comparisons...
This presentation will provide an overview of the waste measurement management at CEA VALDUC from primary producers to the final issue and for the different techniques of measurement from passive neutrons measurements to gamma spectrometry.

*The central laboratory expert in nuclear measurements defines and validates the assay, trains and supervises the operators of other VALDUC laboratories who perform theses measurements and gives them a technical support by a hat line .*

*The central laboratory expert in nuclear measurements performs appraises on wastes and Hold-up on gloves boxes.*

Quality insurance

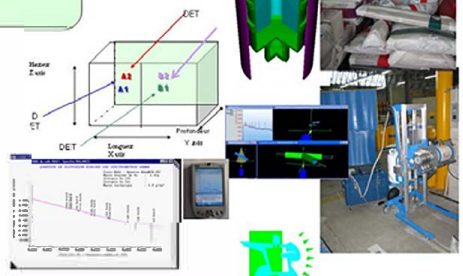
Comparisons organized
for VALDUC laboratories
Internal annual evaluation
on the waste measurements performed
by the others VALDUC laboratories
Annual technical meetings
with the others VALDUC laboratories
performing some measurements
Accreditation
envisaged on waste measurements

The drums of bulk are measured by the primary waste producer using a "*NaI*" gamma detector. This first measurement allows the drum moving to the VALDUC storage before to be measured a second time.

NaI

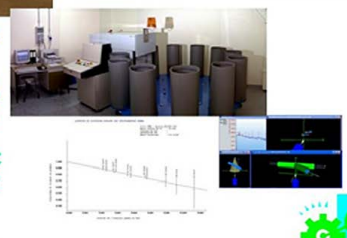
Gloves box pieces are measured by "***Passive neutrons SdC***" counting coupled with isotopic determination by a germanium probe.

Passive neutrons Sdc



Mobile GeHp

The objects, what ever they are, are measured by the technique of the hold-up with a high resolution "***Mobile GeHp***" probe.
The determination of the hold-up mass is based on the methods of infinite energy and modeling. It needs several measurements points and the knowledge of the matrixes and screens.



GeHp carrousel

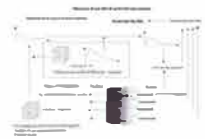
Each waste bag is measured by the primary waste producer using a special "***GeHp carrousel***".
Specifics modelling multi curves are used with each type of matrix.
These bags are put in a drum which will be measured at least to confirm the obtained results.

Total opening case

Madagascar

VVO

Each drum is measured by "***MADAGASCAR***" an automatic gamma assay for waste drums characterisation.
The drums are measured with two high resolution HPGe detectors piloted by NDA2000.
One fixed for the isotopic composition and the second detector, mobile, can perform a segmented gamma scanning of the drum.
The absorption of gamma-rays in the drum is corrected using a transmission source joinsd to the second detector.
The drums which density is higher than 0.5 or containing most metol matrix are measured by the "*VVO*" a drum passive neutrons measurement.



ESARDA 29th Annual Meeting, Symposium on Safeguards and Nuclear Material Management, Aix-en-Provence (France), May 22-24, 2007

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# Evaluation of on-line Single Particle Mass Spectrometry for the screening of environmental swipe samples

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## Introduction

The analysis of environmental samples for tiny particles of nuclear material that reveal the existence of clandestine processes is a powerful tool for Nuclear Safeguards. The principal environmental signature of U enrichment is analysed with high efficiency by secondary ion mass spectrometry (SIMS). However, in cases where uranium particles are present only in very low concentrations among a large surplus of interfering elements, the analysis can become complicated and time-consuming.

## Experimental

On-line Single Particle Aerosol Mass Spectrometers, have been developed as a tool in aerosol research.

→ sample throughput: > 10,000 particles per hour

→ for each particle analysed, the full mass spectrum is recorded, online and in real time

→ size and chemical composition of single aerosol particles can be characterized simultaneously

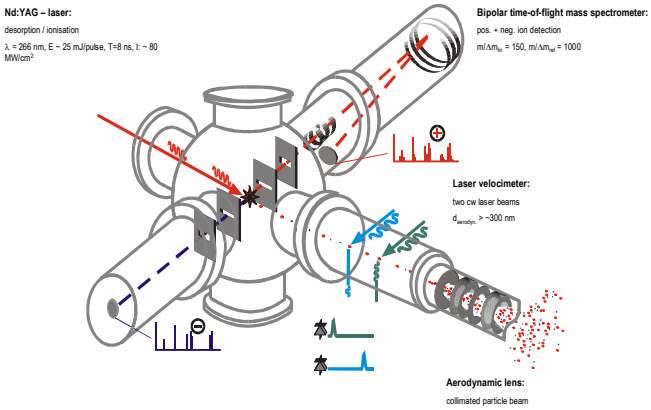


*Fig. 1. Schematic view of the **Single Particle Analysis and Sizing System SPASS**. Aerosols are introduced into the SPASS using an aerodynamic lens system to obtain a well-collimated particle beam with little divergence. Particles are then sized with a two beam laser velocimeter. This information is used to trigger a frequency quadrupled Nd:YAG laser (wavelength λ = 266 nm, pulse length T = 8 ns, pulse energy E = 20 mJ) which desorbs and ionizes the particle, positive and negative ions are mass-analysed in a bipolar time-of-flight (TOF) mass spectrometer. The SPASS has been installed inside a truck to create a mobile unit and has been used in several field campaigns.*

## Samples

a) selected uranium isotopic standard particles

b) four swipe samples, previously analysed by SIMS

## Results

a) for the standards, the difference in enrichment between (natural/depleted), 3% enriched and 10% enriched uranium could clearly be observed in the mass spectra (see Fig. 2), demonstrating the capability of the SPASS for the detection of (undeclared) uranium enrichment activities.

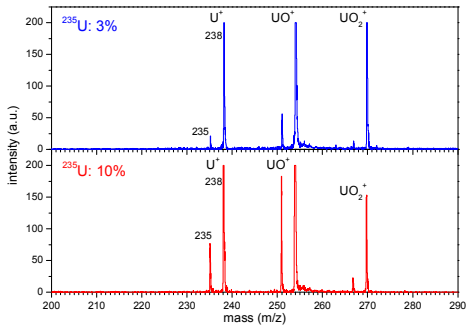


*Fig. 2. Mass spectra of uranium particles with different enrichment.*

b) On the swipe samples, the presence of uranium was detected, together with additional information about other elements and compounds in the individual particles.

Data evaluation is ongoing, a first inspection of the collected swipe spectra revealed signatures from steel or pipes – Al, Fe, Ni, Co, Cr, Pb - together with Ba, Sr, Cd, Li, and sometimes high concentrations of Ti (Fig.3).

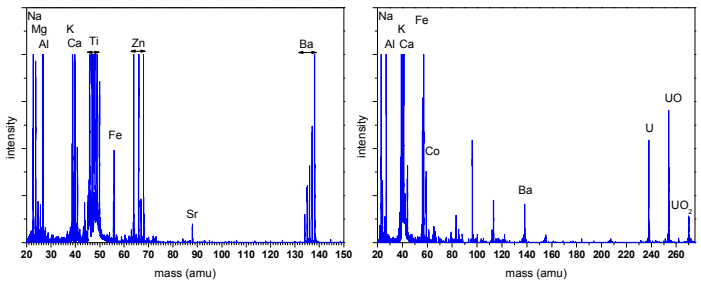


*Fig. 3. Positive mass spectra of 2 single particles samples from swipes. Uranium and other relevant signatures can be seen.*

## Possible applications

→ rapid screening of swipes prior to SIMS

→ identification of characteristic non-nuclear signatures and indicators

## Contact

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