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Session 1

Plenary Session

Managing the tension between risk and burden when considering more efficient safeguards approaches

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Abstract

Stakeholders are agreed in principle that the application of nuclear safeguards should be carried out in as efficient a way as possible. This is clearly of benefit to nuclear operators who are faced with the intrusion and disruption of safeguards inspections and to the inspectorate who must try to make optimal use of scarce resources. Efforts to develop and improve the techniques and methodology for applying safeguards have therefore been and continue to be a strong feature of work in this highly important field. There is of course a balance that must be struck between the sometimes-competing goals of effectiveness and efficiency and neither should be pursued to the exclusion of the other. Due account has to be taken of potential diversionary pathways and the possible consequences of successful diversion of nuclear material, as well as the likelihood of diversion taking place and the burdens that arise in applying nuclear safeguards. As exploitation of nuclear energy continues to attract considerable public scrutiny, a further element is that of perception and the need to maintain and demonstrate a persuasive level of reassurance that nuclear material is not diverted. This paper considers the tension that exists between these factors and what this ultimately means for the credibility of any system of nuclear safeguards.

Keywords: safeguards; nuclear; Euratom; IAEA; plutonium; diversion

1. Introduction

The recent history of nuclear safeguards has been one of strong evolution and development, largely driven and accelerated by the many challenges that have arisen since the early 1990's. This has led not only to new and strengthened IAEA safeguards measures but also to solutions that provide for the more efficient use of inspection and verification resources. Nuclear operators in turn can benefit from these developments, especially where they give rise to a less intrusive and disruptive approach to verification and inspection. For example, it is now widely recognised that the increased use of remote monitoring technology, authentication of direct feed from operator instruments, containment and surveillance measures and approaches that include short-notice, randomised inspections and electronic 'mailbox' delivery of nuclear materials accountancy data, can all provide for the more efficient and effective implementation of nuclear safeguards.

At the same time as considering more efficient approaches, it must be understood that efficiency is not just about reducing costs. It is intimately connected to effectiveness. More efficient safeguards must mean implementing them in such a way that their effectiveness is maintained (even improved), where possible at lower overall cost. Retaining the credibility of the system would mean any *reduction* in effectiveness would have to be fully and openly justified to stakeholders before it could be accepted. Therefore, for any substantial change on the grounds of efficiency it is incumbent upon those responsible for nuclear safeguards to demonstrate that effectiveness would not decline significantly, or

* The views expressed in this paper are those of authors and do not necessarily represent UK Government policy.

otherwise to explain and justify the change. *This requires at minimum a comparison to be made between the costs and effectiveness of an existing approach and that proposed for the future.* To quote a former Director of what was the then Euratom Safeguards Office [1], "To postulate protection at no or reduced cost is considered a misleading if not dangerous illusion".

The costs associated with developing and sustaining a new safeguards regime, including those arising from consultation, peer review, establishing new methodologies and standards, acquisition of new skills, software and equipment, are not inconsiderable and must be factored into such a comparison. The costs of bringing about the change must be readily justified by the gains in effectiveness and efficiency of the new approach. There will also be a need for increased outreach and education and it will be necessary to ensure that the expertise and capability of the inspectorate is maintained. The latter in effect sets a lower threshold to the number of experts required. Inevitably, it also means that there are diminishing returns to be gained from implementing efficiency measures, as the inspectorate must always have a sufficient mass of experts to be able to maintain its knowledge of the many varied and complex nuclear activities and installations subject to inspection, as well as its ability to perform inspections effectively in all these scenarios. The costs to nuclear operators must also not be overlooked when considering new approaches as they can be impacted upon significantly. Any new safeguards approach must remain sympathetic to nuclear plant operational requirements and take account of any consequences that operators might face, especially those that require them to take on new tasks or adopt new or increased responsibilities.

A key question is, then, how can the confidence of stakeholders be retained as 'more efficient' approaches are developed? How can it be demonstrated that effective safeguards will be maintained under a new regime and that the costs of change are justified? In Europe today this is not purely an academic question - the European Commission has developed, and begun implementing, far-reaching changes to its Euratom safeguards mission.

2. Benefits of safeguards in Europe

The authors start from the basic premise that nuclear safeguards are an important and valuable part of the overall regulation of the European nuclear industry. They help to advance public and international confidence in the peaceful exploitation of nuclear energy by providing credible assurance that nuclear material has not been diverted from intended civil uses. It is clear that, globally, it is IAEA safeguards that underpin the non-proliferation regime in providing the all-important assurances that non-nuclear weapon States party to the NPT are keeping to their Treaty obligations. But within Europe we also have safeguards carried out by the European Commission under the Euratom Treaty. It is fair to ask what it is that these do to complement, or add to, IAEA safeguards? The European Commission states in its brochure "Nuclear Safeguards – Europe Remains Vigilant" [2] that "(Euratom safeguards) allow uniform and independent safeguards in all EU nuclear installations. For a very reasonable price, it has been possible to manage the development and peaceful use of nuclear energy for over half a century in Europe".

Although the Euratom Treaty is not a non-proliferation treaty, it is evident that Euratom safeguards have the same basic objective as those of the IAEA – that is, to detect (and thereby deter) the diversion of civil nuclear material from peaceful use. In short, Euratom safeguards complement those of the IAEA and have also served to:

- Provide credible assurance that diversion is not taking place anywhere in the EU.
- Ensure safeguards are applied to all civil nuclear installations in the EU, not just to those in the non-nuclear weapon states.
- Facilitate the use of nuclear energy and related trade and service provision, e.g., through safeguards clauses in cooperation agreements.
- Ensure a distortion in competition does not arise between the EU nuclear weapon states and non-nuclear weapon states.
- Advance standards of nuclear materials accountancy within the EU.
- Promote enhanced, regional approaches to systems of accountancy and control.

- Provide the services of a State System of Accountancy and Control for all Member States (itself an efficiency because of economies of scale and a reduction in need for duplication at the State level).
- Support the international safeguards mission of the IAEA and hence, also the non-proliferation regime.

2.1 Supporting the IAEA

The strong, intrinsic support of the EU for the non-proliferation work of the IAEA has, in the past, been given practical embodiment through the safeguards activities of the European Commission. The partnership arrangement with the IAEA that underpins this has been characterised as the most advanced form of such cooperation so far achieved. Joint inspections, sharing of analytical samples and shared equipment and training have all helped the IAEA to limit its resource expenditure in the EU and apply the savings realised to areas of greater proliferation concern or challenge. It is the view of the authors that this cooperation and partnership has been a recognised and commendable feature of European commitment to the worldwide non-proliferation regime. That commitment and partnership is, in turn, essential to the credibility of the efforts of EU Member States that have been at the forefront of helping to reduce proliferation risk.

2.2 Enhanced credibility of a regional system

Credibility in the proper use of civil nuclear material in Europe is considerably enhanced by the existence of a regional system of safeguards. This provides oversight and assurances that are not only independent of the nuclear industry but are also independent of any one Member State. With much discussion in the IAEA [3] and elsewhere about possible multinational arrangements for nuclear activities, especially those concerning the production of enriched uranium and reprocessing of plutonium, an increased role for regional approaches to nuclear materials accountancy and control may become a renewed feature of the global safeguards agenda. Such regional safeguards systems could, for example, complement multinational commercial arrangements aimed at building additional confidence in the peaceful exploitation of nuclear energy. Europe has served as both pioneer and model in both respects and this has helped foster confidence in all of our civil nuclear activities and commitments, including those entered into by the EU nuclear weapons states related to disarmament. The latter are especially important as the commitments concerned include a moratorium on the production of fissile material for weapons purposes. An additional example, arising from the UK's 1998 Strategic Defence Review, is the substantial amount of former weapons related material that became subject to Euratom safeguards in the UK [4]. At the same time, the UK committed to restrict withdrawals of nuclear material from safeguards to small quantities insufficient for nuclear explosive purposes. It is a key fact that Euratom safeguards remain the only means of providing the public, other Member States and the wider international community with independent assurance that such important disarmament related commitments and initiatives are being observed. It may be a coincidence that the NPT Review Conference is taking place in New York as this paper is delivered but it should serve as a salutary reminder that Euratom safeguards do not take place in a vacuum.

Today, there is also a widespread perception that the risks of misuse of nuclear material have been heightened, whether through proliferation or by acts of terrorism. At the same time the world appears poised upon the verge of a nuclear renaissance, with a growing recognition that nuclear power cannot be ignored as a means of meeting future energy needs. It is, perhaps, more important than ever that the hard-earned, high-credibility of Euratom safeguards should be carefully maintained, not least as a key means of protecting our nuclear industries from any suspicion of misuse of the sensitive materials in their custody. It is the view of the authors that it remains highly desirable that Europe, with its long nuclear history and open and democratic institutions, should continue to set standards of excellence in nuclear safeguards and strive to serve as a model for a strong regional approach and enhanced cooperation with the IAEA. However, it is important to stress that this does not mean that no change can ever take place, only that it must be approached in an open and transparent way and that great care should be taken to subject new Euratom safeguards approaches to peer review and test. This is essential in order to ensure that they have a sound basis and that there is no loss, indeed no scope for perception of loss of effectiveness and ultimately, credibility in the European safeguards system.

3. Developing safeguards – 'Better Regulation'

It is, today, widely acknowledged that regulation should be fair, effective and affordable and enjoy a broad degree of public confidence. With this in mind, the UK's Better Regulation Task Force [5] devised the following five Principles as a toolkit for measuring and improving the quality of regulation and its enforcement:

- Proportionality
- Accountability
- Consistency
- Transparency
- Targeting

These Principles set the context for dialogue between regulator and stakeholders and are intended for use whenever new proposals are being considered or existing regulations evaluated. The UK Government endorsed and integrated these Principles into its Guide to Regulatory Impact Assessment [6]. Better Regulation is a cross cutting issue that will be an important theme of the upcoming UK Presidency. The European Commission has taken steps to give these Principles life through its Action Plan on Better Regulation and recently committed itself to produce impact assessments for all new proposals adopted by the Commission in 2005 that are listed in its annual work programme. Such impact assessments should conform to the Commission's own communication, Com (2002)276 final, covering what the Commission considers should be included and covered in a high-quality impact assessment.

In essence, the five Principles should be borne in mind by regulators whenever devising, implementing, enforcing or reviewing regulations. Broadly, it is a good indicator that a regulator is acting in accord with Better Regulation if they take the following approach:

3.1 Proportionality

✓	Ensure that regulatory and enforcement regimes are proportionate to the problems or risks posed.
✓	Identify, minimise and justify the compliance costs imposed.
✓	Take an educational rather than punitive approach to enforcement where possible.

3.2 Accountability

✓	Have clear standards and criteria against which they can be judged.
✓	Publish proposals and consult all those affected before decisions are taken.
✓	Clearly explain how and why final decisions have been reached.
✓	Have clear lines of accountability to Ministers, Parliaments/assemblies and the public.

3.3 Consistency

✓	Work together with other relevant regulators in a joined-up way.
✓	Ensure that new regulations take account of other existing or proposed regulations, whether of domestic, EU or international origin.
✓	Ensure that regulations and standards are implemented and enforced fairly and consistently.

3.4 Transparency

✓	Ensure the objectives and need for regulation are clearly defined and effectively communicated to all interested parties.
✓	Keep regulations simple and user-friendly.

✓	Undertake effective consultation before proposals are developed, to ensure that stakeholders' views and expertise are taken into account.
✓	Give stakeholders sufficient time and information to be able to respond to consultation.

3.5 Targeting

✓	Ensure regulation is focused on the problem and side effects are minimised.
✓	Focus enforcement on those whose activities give rise to the most serious risks.

3.6 Implications of Better Regulation for nuclear safeguards

It is clear that the exploitation of nuclear energy attracts and will continue to attract considerable public interest. As with safety and pollution and other areas of great public sensitivity, it is essential that a high level of confidence can be maintained. In the current climate of heightened concerns about proliferation and terrorism, it is incumbent upon those States with significant nuclear activities to be able to provide assurance that sound arrangements are in place that will prevent the misuse of nuclear material. As discussed, Euratom safeguards are one of the key means of demonstrating that this is so throughout the European Union. It should be plain that observing the principles of Better Regulation with respect to Euratom safeguards is highly important to the maintenance of public and international confidence in the custodianship of nuclear material within the European Union.

It is then essential that Member States, the IAEA and nuclear operators are fully consulted on the development of any new approach and that their views and expertise are taken properly into account before proposals are published. It is also crucial, not least in order to ensure proportionality and accountability, that a comprehensive risk assessment is undertaken and that this is published and acquires the clear support of stakeholders. As the recent Hampton report "Reducing administrative burdens: effective inspection and enforcement" [7] produced for HM Treasury makes clear, regulators should use comprehensive risk assessment to concentrate resources on the areas that need them most and all such risk assessments should be made public [8]. The Hampton report further underlines [9] that the practical consequence of a lack of comprehensive risk assessment is not only that unnecessary inspections may be carried out but also that *necessary* inspections may not be carried out. It is regrettable that the European Commission did not take a Better Regulation approach when developing its new ways of implementing Euratom safeguards but recent indications that steps are being taken to rectify the situation, e.g., by seeking the views of Member States, are welcome. We look forward to the furtherance and broadening of such discussion but to be truly credible this should be undertaken with a view to obtaining collective agreement with Member States on how best to implement Euratom safeguards in future.

4. The European Commission's new approach to safeguards

The foregoing serves as context (and theory) for ongoing efforts to develop and implement a new Euratom safeguards approach. The European Commission have informed Member States of the general outline for its new approach to implementing Euratom safeguards. Our understanding is that the new approach is based on principles that have been developed and used internally by the Commission to determine that a considerable, even radical change in emphasis was needed for its safeguards mission. This change is reflected in terms of (some might say predicated on) a lower inspection frequency, much less use of physical verification of nuclear material, increased audit of operator arrangements for nuclear materials accountancy (NMA) and checks of performance against those arrangements. In summary, the principles upon which the Commission appear to have founded their new safeguards approach are the following:

- Nuclear operators are considered the first line of defence against diversion.
- This increased responsibility of the operator results in Commission inspection activities based primarily on audit of operator materials accountancy and management systems.

- Euratom Treaty safeguards are a 'control of conformity' as distinct from a 'control of final use' that the Commission consider the basis for comparable IAEA inspections[1].
- Systematic inspections based on criteria from the international non-proliferation regime are therefore no longer necessary (and timeliness is not a factor).
- Level of physical verification is instead to be based on the (low) probability of diversion by the operator.
- Containment and surveillance techniques are linked only to security of nuclear material.

4.1 Control of conformity

The Commission has fully acknowledged that the task of ensuring that operator declarations are correct is the focus of their activity under chapter 7 (safeguards) of the Euratom Treaty. This has been characterised by the Commission as a 'control of conformity'. The Commission have though said that the (primary) objective of this control is to ensure that *operators* have systems in place that would enable them (the operator) to detect the diversion of a small amount of material, with a high probability, in a short time[2]. This does not appear to match the safeguards task placed on the Commission by the Euratom Treaty, e.g., Article 82 of which says that *Commission inspectors* shall be responsible for obtaining and verifying the records that operators keep to account for their production or use of nuclear materials.

The Commission have also referred to theft and diversion as if they are interchangeable terms and could both arise from the actions of individuals or groups inside or outside the operator's organisation. As we show in the next section this appears to be to misunderstand what diversion is and is not. We believe that taking this as a basis for a new approach to implementing nuclear safeguards risks creating a flawed system.

The Commission have further explained that there are three elements to their conformity control. These are listed below with an outline of the activities that we understand the Commission intend to undertake in each case:

- Compliance control – examination of operator declarations and organisation of NMAC systems to ensure they comply with the requirements of chapter 7 of the Euratom Treaty and the related Commission Regulation.

- Performance control – evaluation of the capability and adequacy of operator NMAC systems and management arrangements to achieve the objective (i.e., detection of a small material loss with a high probability in a short time), using a combination of audit methods and expert judgement.

- Credibility control – physical verification of nuclear material to detect diversion by the operator. This would be planned around the much lower probability of operator misbehaviour (as compared to what the Commission perceive to be a higher probability of theft). Consequently physical verification would be based on a random selection of only a small portion of items/materials.

It is clear from this that the Commission consider it to be their responsibility to ensure that operator NMA systems are capable of detecting theft of nuclear material. Operators and national physical protection authorities would be best placed to discuss this with the Commission but it must be clear that NMA systems do not have unlimited resolution and particularly for bulk handling plants, could not be expected to detect the removal of a small quantity of material (see the next section regarding the role of physical protection and an in-depth approach to safeguards). Nor are NMA systems designed to provide the sort of fast response that would be necessary if they were to be useful in actually helping prevent theft. It seems though that the compliance and performance controls are intended mainly as the means of achieving this sort of end.

It is only the so-called 'credibility control' that corresponds directly to the safeguards task of the Euratom Treaty – i.e., verifying nuclear materials accounts to ensure no diversion. The basis (low probability of operator misbehaviour) of the credibility control suggests to us that the risk assessment

[1] As mentioned earlier in our view both systems of safeguards have the same basic objective – that is to detect and deter the diversion of civil nuclear material from intended, peaceful uses.

[2] It is unclear what the terms 'small, high, short' mean as they have yet to be defined.

that underpins the new approach has focused almost exclusively on the *likelihood* of diversion taking place. Furthermore, it seems that 'self-policing' by the operator is both integral and important to the assurances that could be provided under the Commission's new Euratom safeguards approach. The following section therefore outlines our examination of the fundamental questions we believe relate to such a risk assessment and why this leads us to question the foundations of this new approach.

5. Safeguards & a comprehensive risk assessment

According to the entry 'Safeguarding Nuclear Materials' on the Commission's website [10], "Chapter 7 of the Euratom Treaty establishes a safeguards system designed to ensure that civil nuclear materials are not diverted from the civil uses for which they are intended, as declared by operators of nuclear installations". The Commission safeguards brochure 'Europe Remains Vigilant' [2] further states that 'controls concerning plutonium and highly enriched uranium have the highest strategic value since these materials are used for making nuclear bombs' and that 'the main objective of the (inspection) activity is to detect a risk of diversion of nuclear material for other purposes such as making a nuclear bomb, or illicit trafficking'. In a nutshell these statements seem to the authors to summarise correctly what is expected of Euratom safeguards. In reaching the conclusions here we have considered recent thinking about risk, especially as set out in the paper 'Smart Regulation and Risk Management' by William Leiss [11] and in the Hampton report [7].

5.1 Diversion of nuclear material

Before considering risk in more detail, we should be clear about what diversion is, and what it is not. This is essential in order to establish a sound foundation for developing or reviewing a system of safeguards. There should, for example, be no confusion between the objectives of nuclear safeguards, nuclear safety and physical protection, a point well made in the aforementioned Commission safeguards brochure [2]. It is true that physical protection and nuclear safeguards both have the objective of reducing the risk that nuclear material will be misused – the main considerations of misuse are firstly for a nuclear explosive device and secondly, a radiological dispersal device (RDD or 'dirty bomb'). Such misuse could in theory result from terrorist action at a nuclear site or from the theft or diversion of nuclear material. It is clear that physical protection measures are targeted at the first two of these threats and, what we in Europe call nuclear safeguards, only at the third, i.e., the target of Euratom Treaty safeguards is the threat of diversion.

It is incontrovertible that diversion of nuclear material would be an act undertaken and authorised by those that had it in their possession and control, i.e., the nuclear operators (whether or not with State collusion). The deliberate, *unauthorised* removal of nuclear material is theft, not diversion, whether carried out by an insider or external individual or terrorist group. In other words, the operator cannot be considered the first line of defence against acts of diversion, as the operator would, by definition, be party to and directly responsible for such an act. ***Safeguards are therefore the first line of defence - they are the only independent means of detecting and thereby deterring the diversion of nuclear material.***

Such analysis suggests that a Euratom safeguards regime that focuses or relies on audit of operator systems will have a greatly reduced capability of detecting diversion of nuclear material and a correspondingly reduced credibility. It should be clear that NMA system audit and the compliance and performance controls described above can be no substitute for the physical verification of nuclear material in determining the accuracy of operator accounts and hence detecting diversion. Only the independent physical measurement of nuclear material can make sure (verify) that operator accounts reflect reality and are therefore correct and do not attempt to conceal diversion.

This is not to argue that some form of audit or assessment of operator NMA systems and performance against them has no role in an in-depth approach to implementing nuclear safeguards (e.g., as developed for and implemented at large scale reprocessing plants and other facilities that handle unirradiated direct use material). Such 'safeguards-in-depth' use several layers of access to information, equipment and its output and the nuclear material itself, to provide confidence that there has been no diversion. Direct physical verification of the nuclear material nonetheless remains fundamental to the detection capability. Greater understanding of operator NMA systems might also have a role to play in improved, more sophisticated safeguards at other kinds of facility. In addition,

NMA performance audits could provide an element of quality control in helping to ensure and maintain good standards of accountancy. Of course, just how much could or should be done in terms of such system audit would need to be a function of the factors mentioned in this paper, i.e., a comprehensive assessment and comparison of safeguards risks and effectiveness of tools available to combat them.

5.2 Physical Protection or Safeguards?

Physical protection measures are of course intended to protect nuclear material and nuclear installations from malicious acts, such as sabotage or theft. They involve making installations and material secure through measures such as access checks and controls, barriers, intruder alarms, surveillance, guards and response forces, etc. Some of these clearly act as visible deterrents. They must also operate in real or near real-time to be effective. Detecting that material has already gone missing, the role of safeguards and one of the roles of nuclear materials accountancy, is too late in physical protection terms[3].

Nuclear safeguards, by contrast, do not themselves prevent diversion but are instead designed to detect that material may have been diverted and be a trigger for further investigation and action. Unlike physical protection, safeguards therefore provide an alarm some time after the fact but in so doing can provide deterrence against diversion. They will of course only provide worthwhile deterrence if there is a significant chance that diversion will in fact be detected and in a timely manner. Safeguards have, then, to be designed around a reasonable probability of detecting diversion (usually of a significant amount of nuclear material) in a timeframe commensurate with the risk presented, i.e., they should take account of the time it would take realise the potentially catastrophic consequence of diversion. This in turn means that safeguards approaches must, as a primary consideration, take account of material type and form and what constitutes a significant amount of such material. Our understanding of the risk assessment underlying the Commission's new approach to Euratom safeguards (based on the principles and 'credibility control' referred to earlier) is that little account has been taken of the nature of the nuclear material concerned and that this gives rise to a system with a low probability of detecting diversion across all material types, even those referred to in the Commission's own safeguards brochure [2] as having the highest strategic value.

5.3 Elements of risk

There are four elements that can be considered as integral to any analysis of the risks concerning the diversion of nuclear material. These are:

- Likelihood of an operator diverting
- Diversionary opportunities (pathways)
- Difficulty of obtaining fissile material
- Consequence of diversion

Of these it is immediately apparent that the consequence of successful diversion could be huge, especially if it resulted in the detonation of a nuclear explosive device. Even the possible acquisition of a nuclear bomb or credible threat of use of such a weapon would have the most profound consequences. We cannot assume that any of the first three factors have a zero probability either and given this, the authors regard the final factor as key. In other words, where consequences are potentially so immediate and dire there is a clear rationale and imperative for maintaining a high level of vigilance in order to detect and deter diversion, even if the probability of diversion happening can be considered low. This is in keeping with the issue of proportionality referred to above under the banner of Better Regulation – in such instances a risk-averse approach is clearly justified as well as been a matter of public expectation.

That said the difficulty, diversionary pathways and likelihood must not be ignored. Within the EU it would be fair to conclude that the risk of diversion is at present relatively low. It would also seem to us

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[3] If a theft of nuclear material were only detected by an operators NMA system then the physical protection regime would be seen to have failed – if such a situation arose the NMA system would serve to trigger investigations into how those defences had come to be evaded.

to be reasonable for the European Commission to be able to conclude for itself that there are at present no indications of clandestine nuclear production or processing facilities in the EU. This would in turn lead in the direction that safeguards applied to indirect use materials and irradiated direct use materials (many of which are unattractive for an RDD, especially compared to other radioactive materials and all require substantial further processing to obtain fissile material), need not be driven by the same high detection probabilities that should be applied to the more strategic, unirradiated direct use materials (plutonium and HEU). The conclusion must again be that material nature and amount are the most critical factors in the safeguards risk assessment.

5.4 Direct vs. indirect use material

It is clear that the difficulty of obtaining fissile material usable in a nuclear bomb is very low for direct use material and the consequence of diversion could be relatively immediate, huge and catastrophic. It is therefore the conclusion of the authors that, in order to maintain the credibility of Euratom safeguards and public and international confidence in EU nuclear activities, the safeguarding of unirradiated direct use materials must continue to be based primarily on the independent use of direct physical verification measures. It seems clear that, given the risks, these safeguards should be designed to achieve a high probability of detecting the diversion of a significant amount of material in a timely fashion. This in turn means that (e.g., for separated plutonium in storage) the most efficient approach will continue to involve use of mature safeguards technologies, such as containment & surveillance to maintain verification knowledge once it has been obtained by safeguards inspectors through the direct physical measurement of the material.

A lower probability of detection might be considered as providing suitable assurance and deterrence for irradiated direct use and indirect use materials. Whilst the eventual consequence of successful diversion is still high, the likelihood of diversion is low and the difficulty of obtaining fissile material is high. Drawing a conclusion about the absence of clandestine nuclear activities in the EU and taking account of the time to not only obtain fissile material but also to construct the facility or facilities needed to process the diverted material and undertake that extraction, can provide a sound basis for more effective and efficient Euratom safeguards for these materials. It is notable that the IAEA and its Member States and ESARDA have carried out much work in this field, in particular as part of the development of integrated safeguards. Efforts to design more effective and efficient approaches to implementing Euratom safeguards would seem to be best served by benchmarking against and making as much use as possible of the lessons learnt during that work.

6. Conclusion

The authors fully acknowledge that safeguards must change to keep up to date with new techniques and technology. We have mentioned some of these in passing, e.g., random short notice inspections, remote monitoring, mailbox arrangements, all of which can help realise efficiency gains and have our support. All in all we believe that such developments provide a sound and arguably overdue basis for change in the implementation of Euratom safeguards.

But we still need to answer the questions set at the beginning of this paper - how can the confidence of stakeholders be retained and how can it be demonstrated that effectiveness will be maintained and that costs are justified? To achieve this, the authors firmly believe that the changes must be derived from a risk assessment which is comprehensive, accessible and acceptable to all those with a stake in Euratom safeguards. As this paper makes clear, for safeguards purposes it is nuclear material type and form that should be the most critical aspect of any such risk assessment. We also believe that a comparative methodology is needed, in order to demonstrate that the effectiveness and costs of any new approach have been properly evaluated against the system it is designed to replace. The principles of Better Regulation are considered to provide an excellent framework for undertaking this process and if followed, should ensure that the credibility of Euratom safeguards is protected and secured for the future.

It is worth repeating here one of the conclusions reached by William Leiss [11] that “without a credible methodology in place for the evaluation of the key criteria (efficiency and effectiveness), we are unable to make defensible judgements about optimal policy mixes for realising specific objectives.

Advocating the changing of regulatory structures in the absence of such a methodology is a case of the blind leading the blind".

The challenge now for Euratom safeguards is to continue the debate and discussion that has really only just begun, in order to reach a consensus on the changes that can safely be made, that will improve efficiency and preserve effectiveness of those safeguards.

7. References

[1] W. Gmellin; Safeguarding Plutonium: A Continuing Challenge; European Commission Euratom Safeguards; 2000

[2] Nuclear Safeguards – Europe Remains Vigilant; European Commission DG-TREN; 2003 (to be found on the Europa website)

[3] Multilateral Approaches to the Nuclear Fuel-Cycle: Expert Group Report Submitted to the DG of the IAEA; INFCIRC/640; 2005

[4] Communication Received from the United Kingdom of Great Britain and Northern Ireland; INFCIRC/570; 1998

[5] Better Regulation Task Force; Principles of Good Regulation; www.brtf.gov.uk; 2003

[6] Better Policy Making: a Guide to Regulatory Impact Assessment; Cabinet Office; www.cabinet-office.gov.uk (links to the Regulatory Impact Unit and scrutiny); 2003

[7] P. Hampton; Reducing administrative burdens: effective inspection and enforcement; HM Treasury; www.hm-treasury.gov.uk/hampton (the Hampton report); 2005

[8] See recommendation 1 and paragraphs 2.22, 2.32 and 2.45 of the Hampton report.

[9] Paragraph 11 of the executive summary of the Hampton report.

[10] Safeguarding Nuclear Materials; European Commission DG-TREN; 2004; Europa website.

[11] W. Leiss; Smart Regulation and Risk Management; prepared for the Canadian Privy Council Office; 2003; www.smartregulation.gc.ca

ESARDA

A new agreement

for new opportunities

ESARDA

The context

- **Reinforcement of the control of nuclear materials**
- **Significant changes in the implementation of safeguards in the European Union**
  - **implementation of Safeguards and the Additional Protocol in the European Union**
  - **evolution of the European Commission's policy on nuclear safeguards**
- **Evolution of the street perception by the public, national administration and political levels**

ESARDA

a new agreement

the targets

- A new agreement in place by the end of 2004
- Be simpler and much easier to authorize/sign
- As free from legal liabilities as possible
- Keeping from the past
  - the existing partners
  - the secretariat provided by JRC
- Encourage the participation of new members
- More efficient and effective organizational management
- More rapid
  - arrangements for member joining/leaving
  - arrangements for initiating/closing working groups

ESARDA

New agreement : The purpose (1/2)

- To improve the quality, the efficiency and cost-effectiveness of nuclear material safeguards, including nuclear material management and accountancy
- To seek the views of all those concerned in the application of safeguards so that problems preventing their effective, efficient and economic application may be identified and solved
- To facilitate collaboration on R&D in safeguards and related fields and its application

ESARDA

New agreement : The purpose (2/2)

- To propose R&D programmes in the light of identified safeguards and nuclear non-proliferation topics which require investigation and to facilitate collaboration
- To take benefit and stimulate synergies with other verification regimes and technologies
- To increase the understanding of safeguards by improving communications with the public and other experts.

ESARDA

New agreement : The activities (1/2)

- **Harmonisation and promotion of the R&D programmes or other works of the members, by the exchange of information and assistance on the personnel and technical levels and by the joint execution of these programmes or parts thereof**
- **Providing a forum for consultation involving operators, regulators and researchers**

ESARDA

New agreement : The activities (2/2)

- **Promoting the organisation of workshops, conferences, symposia and training on general or specific safeguards related topics**
- **Dissemination of the results of the programmes and actions.**

ESARDA

New agreement : The parties

Organisations or corporate bodies within the European Union which shall have a legal status and meet the following criteria:

- **Provide high-level R&D in line with the purpose of ESARDA**

or

- **Be an operator of a nuclear installation subject to nuclear material safeguards**

or

- **Be involved in safeguards as an agency or a regulator.**

ESARDA

New agreement : Associated and individual members

- associated organisations or corporate bodies may cooperate in ESARDA activities
- they shall meet all criteria set for the Parties except that of belonging to the European Union.
- Individuals may also cooperate within the scope of work of this Agreement as individual members. Their cooperation shall be proposed by an existing Party.

ESARDA

New agreement : accession of a new party

- **The accession to the ESARDA Agreement of any new Party shall require a simple majority vote of the Steering Committee.**
- **The membership will become effective on the date the new Party signs the Agreement.**

ESARDA

New agreement : Steering Committee

- **Made up of all Parties of the Agreement, each Party having one vote.**
- **Associated and individual members may attend the meetings but shall have no voting rights.**

The Steering Committee discusses the general ESARDA policy, takes resolutions and decisions to be implemented by the Executive Board.

ESARDA

New agreement : Executive board

- **Made up by the representatives of five Parties, elected by the Steering Committee for a period of four years, and by the President and the Vice-President, also elected by the Steering Committee for a period of two years.**

- **Every two years two or three seats of the Board are vacated alternately.**

The Agreement is implemented by the Executive Board.

ESARDA

New agreement : Working groups

- **Working Groups to discuss or execute agreed research, development or application programs, or to examine specialized technical subjects**
- **WG are created by the Executive board**
- **Each Working Group shall provide an annual activity report**

ESARDA

The status

- The new ESARDA agreement is in force since 15 March 2005
- The first members are :

Commissariat à l'Energie Atomique (CEA)	Electricité de France (EDF)
Ente pour le Nuove Tecnologie l'Energia e l'Ambiente (ENEA)	Statens Kärnkraftinspektion (SKI)
Säteilyturvakeskus (STUK)	United Kingdom Atomic Energy Authority (UKEA)
Centre d'étude de l'énergie nucléaire-Studiecentrum voor Kernenergie (SCK-CEN)	Wirtschaftsverband Kernbrennstoff-Kreislauf e.V (WKK)
Intstitut de Radioprotection et de sûreté nucléaire (IRSN)	European Atomic Energy Community (EURATOM)

ESARDA

Our challenge

new members

- **In order to achieve its purpose ESARDA needs :**
  - **– the adhesion of new members, to have the best representation of organizations dealing with application of nuclear safeguards in Europe**
  - **– a large cooperation with associated and individual members to benefit from external point of view**

ESARDA

Conclusions

Now that ESARDA has a modern and efficient legal framework

We can focus now on :

- **maintaining and developing our contribution to knowledge dissemination, harmonization and guidelines in R&D of Nuclear Safeguard and related fields**
- **providing a forum of discussions involving operators, regulators and researchers**
- **anticipating new orientations : become a technical think tank for all stakeholders**

Thank you for your kind attention

Nuclear Security at the JRC

Roland Schenkel

Director General (acting)
European Commission, DG-Joint Research Centre

Outline

- Nuclear Security: JRC in FP7
- International Collaborations
- Concluding Remarks

Nuclear Security: JRC in FP7 (1)

Nuclear Security will be a main JRC research area in the 7th European Programme for R&D [2007-2011]. Activities will evolve along five main areas:

- Nuclear Safeguards
- Additional Protocol
- Observatory for Non-Proliferation Compliance
- Combating illicit trafficking and Nuclear forensics
- Radioactivity monitoring

Nuclear Security in FP7 (2)

Nuclear Safeguards:

- Technical support to DG-TREN and to the IAEA
- Increased automation and better tools for information analysis to reduce both inspector workload and the burden on the nuclear industry.
- Innovation and improvements required to implement the new orientation in safeguards, including:
  - verification and detection technologies
  - measurement methods of nuclear material, on-site laboratories
  - development of improved tools for information analysis
  - particle analysis for the detection of undeclared nuclear activities
  - provision of training for operational tasks

Nuclear Security in FP7 (3)

Additional Protocol:

- Overall description of a country's nuclear activities
- Extensive site declarations, including tools for integration of information
- Support to varied inspection requirements, including:
  - off-site monitoring
  - measurement activities inside/outside facility boundaries (e.g., swipe analysis, environmental monitoring)
- R&D objectives are to move towards real-time follow-up and checking of nuclear material transfers based on the use of improved declaration systems, secure communications and integrated information analysis.
- JRC's major contribution will lie in the development and validation of information analysis tools and methodology based on systems analysis.

▶▶ Example: Open Source satellite imagery

Iran: Esfahan nuclear site

Iran: Esfahan nuclear site

2003

Iran: Esfahan nuclear site

2004

Iran: Esfahan nuclear site

2005

Nuclear Security in FP7 (4)

Observatory for Non-Proliferation Compliance :

- Monitor any potential evolution of the fuel cycle
- Analysis of state (or region) capabilities based on description of entire nuclear fuel cycle and safeguards and Additional Protocol information
- Development of a methodology and tools pulling together information from a variety of sources:
  - Safeguards data,
  - Additional Protocol data
  - open source data – status of installations, importation of materials and technology, commercial circuits used, scientific and technical capabilities, person identification/tracking, …

    ▶▶ Examples: Automatic person recognition, Goods tracking

Open Source: person identification & tracking

Automatically identify links and stories associated with entities

Abu Musab al-Zarqawi

Read Wikipedia entry

Abu Musab al-Zarqawi (Eu,en)
Abou Moussab al-Zarqaoui (fr)
Abu Mussab al-Sarkawi (de)
Abu Mussab el Sarkawi (de)
Al Zarqawi (de,it)
Abu Musab al Zarqawi (en,it)
Al -Zarqaoui (fr)
Al -Zarqawi (en)
Abou Moussab Zarkaoui (fr)
Abu Musab al Zarqaui (es)
Abou Moussab Al-Zarqaoui (fr)
Al Zarqaui (es)
Abu Musab Al Zarqaui (es)
Abu Musab Al Zarqawi (it)
Al -Zarqaui (es)
Abu Mussab al Sarkawi (de)
Abu Mussab al-Sarkaui (de)
Abou Moussab al Zarkaoui (fr)
Abu Mussab al-Zarqawi (en,it)
Abou Moussab al-Zarkaoui (fr)
Al Zarkawi (it)
Abú Musab al Zarqaui (es)
Abu Musab al-Zarqaui (es)
Abu Musab Al-Zarqawi (en,it)
Abu Mussab Al Sarkawi (de)
Abu Mussab al Zarkawi (de,es)
Abu Mussab Sarkawi (de)
Abu Musab Zarqawi (en)
Abou Moussab Al-Zarkaoui (fr)
Abu al Zarqawi (it)
Mussab al-Sarkawi (de)
Mussab al Zarqawi (it)
Moussab al-Zarqaoui (fr)
Al Zarkaoui (fr)
Abú Mussab al Zarqaui (es)
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People

Iyad Allawi
Saddam Hussein
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John Kerry
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Ken Bigley
Margaret Hassan
Jack Straw
Simona Torretta
Hoshyar Zebari
Jack Hensley
Jacques Chirac

Organisations

Open Source: goods tracking

JRC Contraffic tool

Brussels plans probe of Chinese textile imports
By Raphael Minder in Brussels
Published: April 24 2005 13:52 | Last updated: April 24 2005 21:58

FT.com FINANCIAL TIMES

- Dramatic surge of textile imports (up to 500% in 1st quarter 2005)
- Investigation to be launched; potential "safeguards" measures

- Uses data-mining techniques in public registers to detect suspicious container movements, reconstructs vessel movements
- Is a unique data resource on global container movements
- Its route-based risk analysis complements other efforts at national level;
- data provided to national customs offices

Overall hit rate 37% based on global routing information alone

- Products concerned: chemicals, shoes, textiles, porcelain, etc.
- Irregularities: anti-dumping, quota-related infractions

Nuclear Security in FP7 (5)

Illicit Trafficking and Nuclear Forensics:

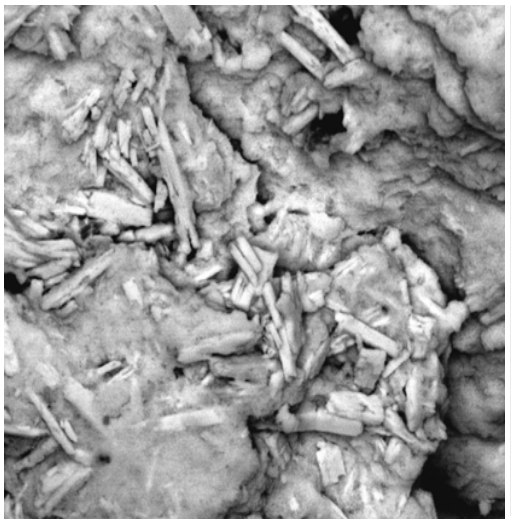
- Increased co-operation with national authorities and relevant international organisations, such as the IAEA or the ITWG
- Strengthen JRC's position as the EU central laboratory for nuclear forensic measurements
- R&D on nuclear forensics to provide more comprehensive info/chain of evidence on the origin of the radioactive materials, and establish appropriate response plans
- Participate in a network of laboratories to qualify the performance of border control equipment, both for hand-held and portal-type instruments
- Corresponding training activities

▶▶ Example: seized material in Rotterdam, 2003

Illicit Trafficking and Nuclear Forensics

Case Study : Find 26 – Rotterdam

On 16th Dec. 2003- 2-3 kg radioactive material was detected in a scrap metal shipment in Rotterdam harbour. The shipment had arrived from a dealer in Jordan. Materials were sent to JRC on 10th March 2004 and consisted of 2 bulk samples and 3 swipes.



Bulk material:

- natural uranium oxides
- uranium content 35 %
- main impurities: Al, Ca, Cr, Fe, Mg, Mo, Na, Ni, P
- lead isotopic composition

Swipes:

- natural uranium, Cs-137, Eu-154 and Am-241 (evidence of nuclear activities)

What information did the analysis yield?

- Impurities point to phosphate rich ores (North Africa, Middle East, USA, South Africa, Brazil)
- Pb isotopic composition (natural) indicates low uranium content in the ore.
- Evidence of nuclear activities
- Corroborated intelligence information that the material had been stolen from stores in Iraq.

Nuclear Security in FP7 (6)

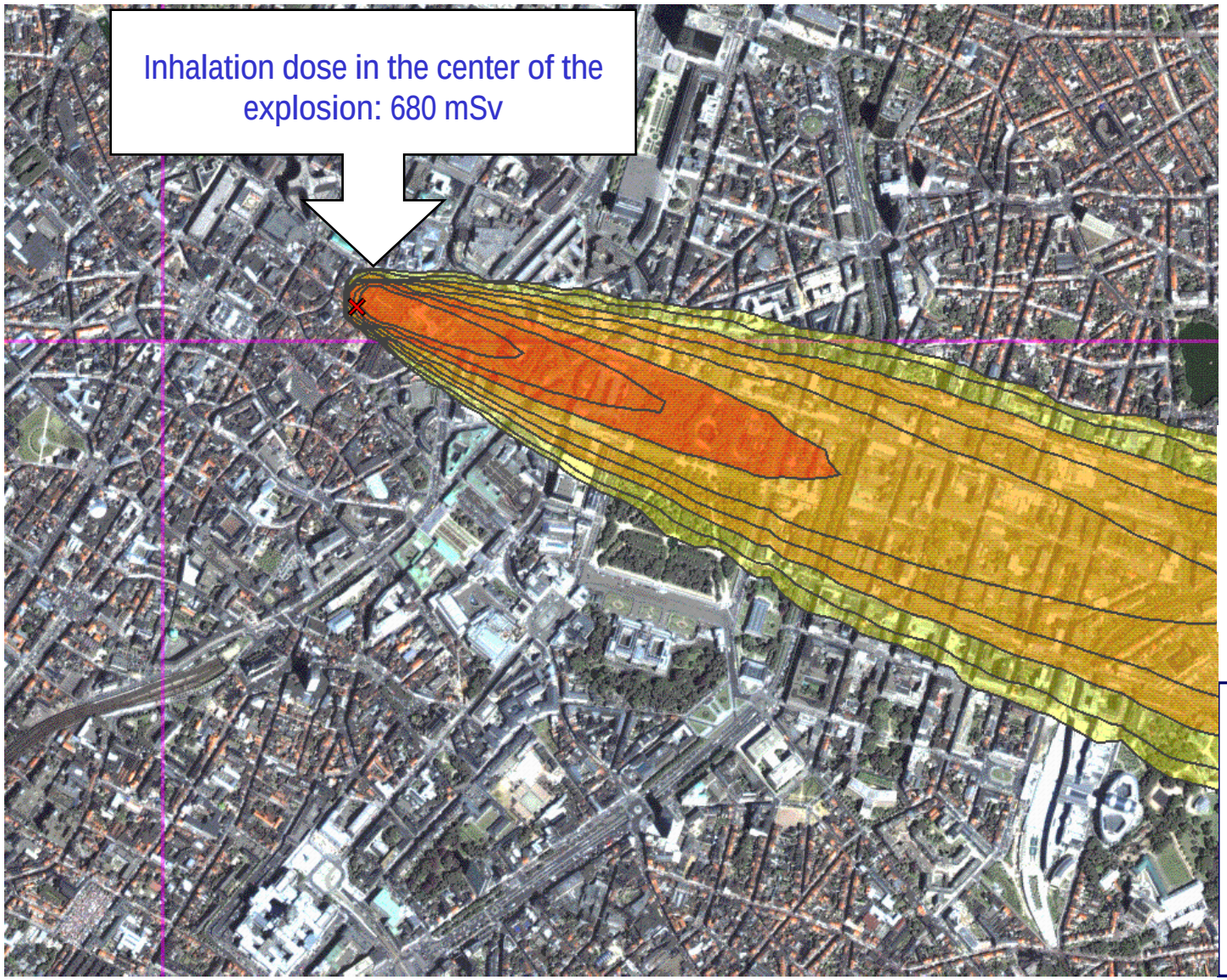
Radioactivity monitoring:

- S/T support to the policy of DG TREN
- Development of trace analysis methods to tasks of verification of radioactive discharges and emissions
- Speciation and migration of actinides in the biosphere
- Carrying out sampling and/or analytical campaigns
- Study of hot environmental particles
- Dispersion modeling of radioactive substances for EU-wide applicability, post event assistance and forensic evaluation capability

►► Example: RDD Scenario

Radioactive Dispersion Devices

The propagation of the radioactive cloud in a city



scale 1:10,000 DMA IKONOS satellite map (wind with 0.5 m/s from 290° north, weather conditions neutral/unstable)

RDD with 2000 Ci ^{60}Co

($7.4 \cdot 10^{13}$ Bq)

1.5 kg conv. explosives

Total for 2 hours

In collaboration with the BfS, Germany

No real explosion!

Joint Research Centre

International Collaborations: Support to the IAEA

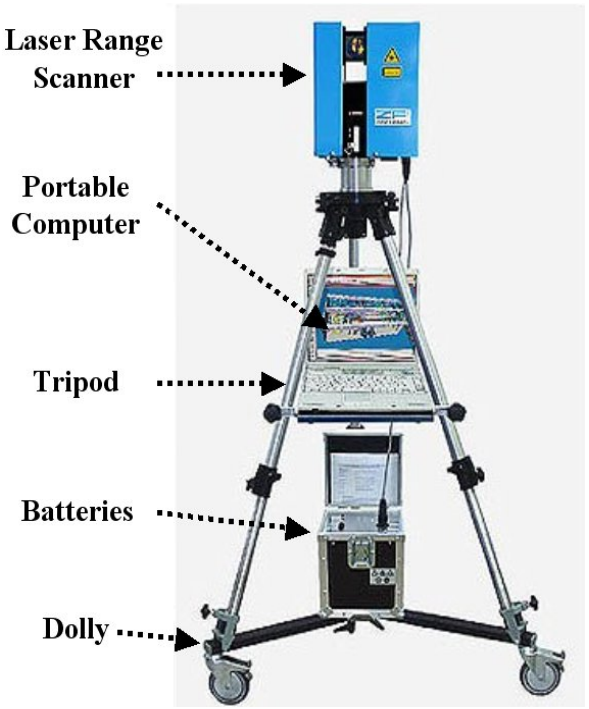
- EC Support Programme started in 1981 [25 years in 2006]
- 3 Institutes involved: IRMM, ITU, IPSC
- Between 15 and 20 active tasks at any time
- Expert consultancy at IAEA Workshops and Technical Meetings:
  - Unattended Remote Monitoring
  - Sealing and Containment
  - Verification of Enrichment Facilities
  - Nuclear Forensics Support
  - Next Generation Surveillance System
  - From Information to Knowledge
  - Reference Materials for DA
  - ....
- Participation in IAEA Coordinated Research Programmes
  - Illicit Trafficking
- Training activities

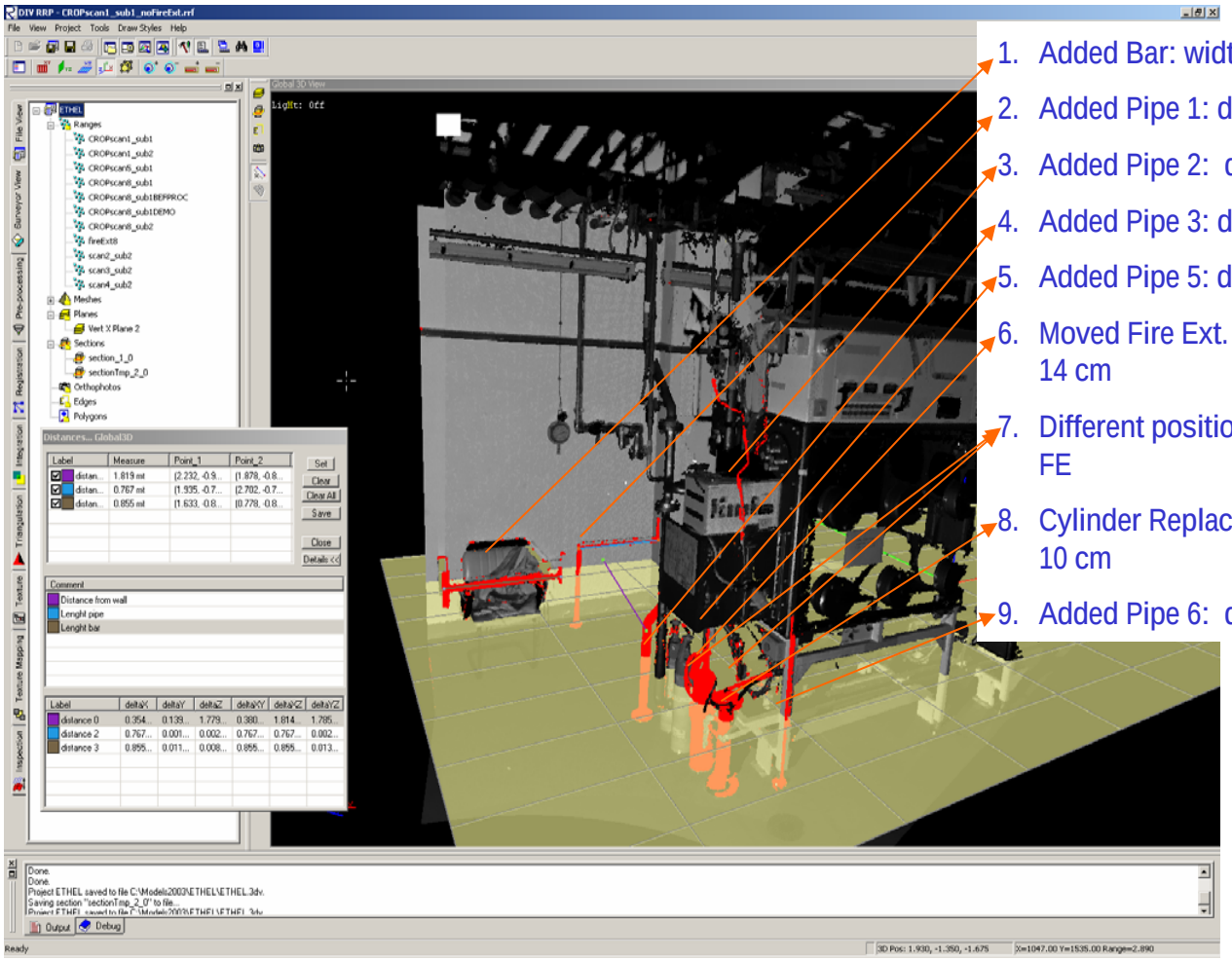
Support to the IAEA (2): DIV

Example: DIV – detecting and reporting the differences

Approach

- Create accurate 3D reference model of the environment
- Compare 3D models (and find differences) from different DIV inspections



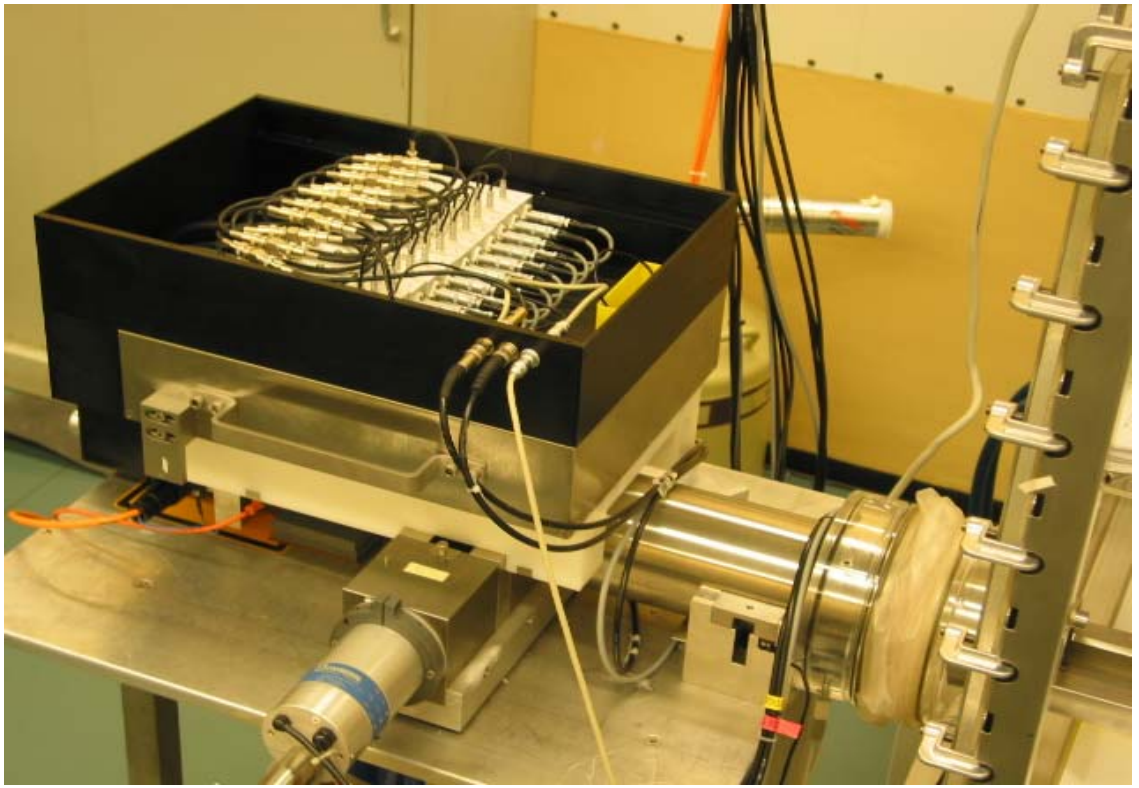


1. Added Bar: width: 5 cm
2. Added Pipe 1: diameter 3 cm
3. Added Pipe 2: diameter 0.6 cm
4. Added Pipe 3: diameter 6 cm
5. Added Pipe 5: diameter 1.2 cm
6. Moved Fire Ext. (FE): diameter 14 cm
7. Different position of pipe from FE
8. Cylinder Replacing FE: diameter 10 cm
9. Added Pipe 6: diameter 1.6 cm

- All distances (e.g., pipe diameters) can be interactively computed for inspection's annotations.

Support to the IAEA (3): OSL

Example: Support to On-site Laboratory at Rokkasho



Cm NCC detector during testing phase at ITU

Input solution:

- Neutron Coincidence Counter ▶ Cm
- HKED ▶ Pu

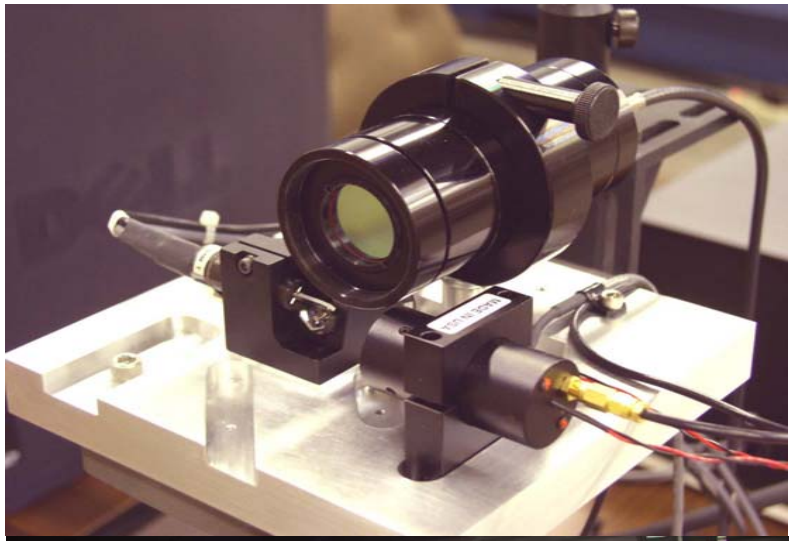
Yields Cm/Pu ratio to deduce Pu in solid samples where only Cm is measured

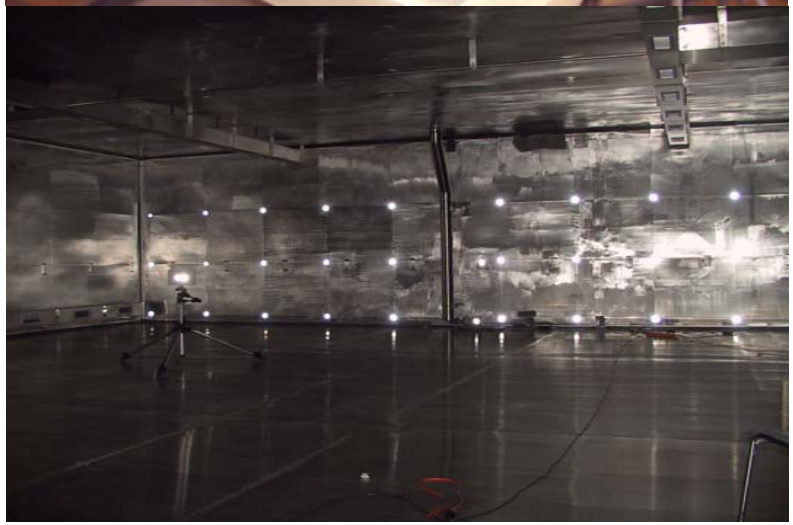
- Development
- Construction
- Testing
- Installation
- Calibration

International Collaborations: DoE – EURATOM

Formal collaboration since 1995

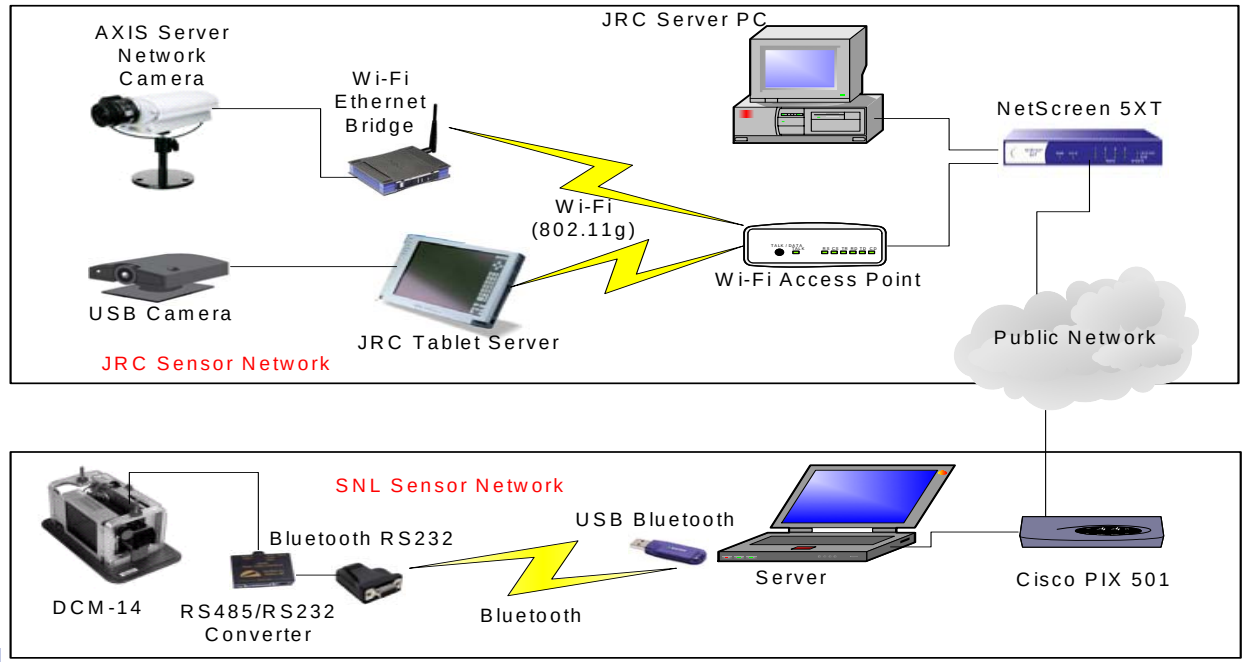
2005 + Projects: Many active tasks involving US national labs and JRC institutes



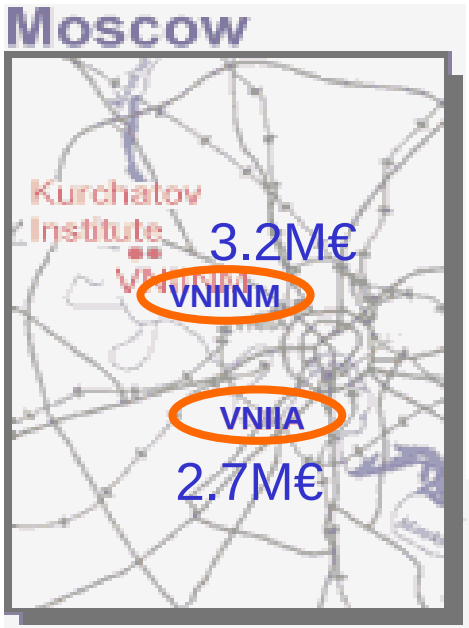


◄ LBIMS: Laser Based Item Monitoring System (with Oak Ridge Labs)

Wireless Data Collection and Secure Transmission (with Sandia Labs)



Int'l Collaborations: TACIS Safeguards Projects (1)



- TACIS: 7 projects / 3 countries – 14.7 M€

  Enhancement of nuclear safety and security by means of:
  - Methodology and Training Center (IPPE-Obninsk and VNIITF- Snezhinsk)
  - Nuclear Material Accountancy and Control (VNIIA, REA, Bochvar and Ulba)
  - Combating illicit trafficking of nuclear materials (INR)

Int'l Collaborations: TACIS nuclear safety & security (2)

New TACIS program 2005-2006

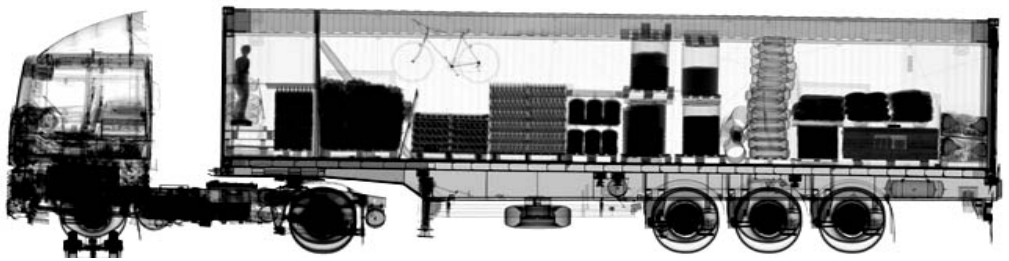
- 15 projects / 7 countries
- Global budget: 32 M€ (Commission proposal)
  - Combating illicit trafficking of nuclear and radioactive materials (Multi-country project)
  - Nuclear Material Accountancy and Control
  - Management of spent nuclear fuel in North West Russia
  - Analytical and methodology support
  - Training
  - Transfer of knowledge and safety culture

International Collaborations: Generation IV

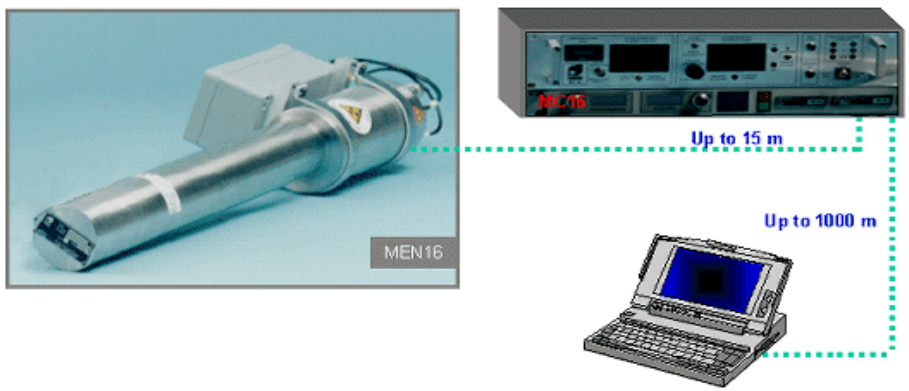
- JRC leads EURATOM participation to GIF (Generation IV International Forum)
- Participation at the GIF's Working Group PR&PP (Proliferation Resistance and Physical Protection)
- PR&PP Workshop at Ispra in 2004
- Systems analysis, risk and threat assessment and analysis: report and review of methodologies and approaches
- Development of analytical methods for new fuel cycles (e.g. pyro-processing, inert matrix fuels)

International Collaborations: Illicit Trafficking

- IAEA and EU Network of labs of testing facilities and border security
  - Qualification of radiation detection equipment
- EURITRACK: detection of explosives and other threat materials in containers
- Joint projects with new EU member states and Romania, Bulgaria, Turkey, western Balkans
- TACIS project with Russia, Ukraine, Georgia, Armenia, Kazakhstan
  - Model border crossing, equip laboratories for nuclear forensic analysis, delivery of portable radiation detectors
- Training for specialists



Combined Technology:
X-Ray Radiography
&
Neutron Interrogation



ESARDA

- ESARDA is over 35 years old.
- ESARDA changed recently its Contract of Collaboration
- ESARDA is actively looking for new members, namely in EU new MS
- Parties were willing to include in the purpose:
  - "… R&D in Safeguards and nuclear non-proliferation topics "

- ESARDA should be aware of its influential and important role in gathering nuclear security competences throughout Europe
- ESARDA Symposium could become the platform for nuclear security research

Concluding Remarks

- Traditional Safeguards is mature. Continuous support and improvements are required to respond to an evolving fuel cycle as well as to keep abreast with changing technologies – "raising the bar"
- Threat of proliferation and Additional Protocol requires a shift in verification philosophy:
  - Nuclear accountancy supplemented by "intelligence" investigation system
  - Change in roles of inspectors towards more qualitative aspects and cognitive capabilities
  - New support and training modules for these new tasks
- Efficient Nuclear Security requires increased international collaboration
- The JRC is ready, both from a technical and international networking perspective, to respond to future nuclear security challenges

Implementation of Safeguards and the Additional Protocol in the European Union

K. Murakami
Director, Division of Operations C
Department of Safeguards
IAEA

Abstract:

The presentation covers the status of work and consultations related to the significant changes that are taking place in the implementation of safeguards in the European Union. The Additional Protocol to the Safeguards Agreement with the Member States and the European Atomic Energy Community (EURATOM) entered into force in April 2004 and the initial declarations for all States party to this Agreement (INFCIRC/193) were submitted to the IAEA by the end of 2004. The review of the declarations and actions to clarify issues related to them is a high priority task to which the IAEA is devoting significant resources. In parallel, the IAEA started carrying out complementary access visits in the Euratom area at the end of 2004. In the ten new EU States the implementation of the Additional Protocol is well advanced in many States and integrated safeguards has been introduced in one State. However, the transfer of safeguards in these new EU States to INFCIRC/193 will present administrative challenges for which the IAEA has been making preparations. In addition the role and activities of EURATOM will change significantly and the technical, policy and resource issues connected with this new situation will need to be addressed in close consultation and cooperation between the IAEA, Euratom and the Member States of the EU. Two issues, among others, that need to be addressed are the review and re-affirmation of equitable and predictable financial sharing arrangements between the IAEA and EURATOM, and the establishment of arrangements to allow the IAEA to schedule and conduct inspections without EURATOM inspectors.

IAEA Safeguards in the European Union

Kenji Murakami
Director, Division of Operations C
Department of Safeguards

IAEA
International Atomic Energy Agency

Overview

I. Consequences of Euratom's Proposed Changes.
II. Additional Protocol Implementation.
III. Global Safeguards Issues.

IAEA

Euratom's Proposed Changes - Milestones

- 16 December 2004
  - First IAEA/Euratom meeting
    - Euratom's general plans described
- 3 February 2005
  - IAEA/Euratom technical meeting on proposal for reactors
- 15 February 2005
  - IAEA briefed EU Member States on consequences of Euratom's proposals
- 22 April 2005
  - Letter from DG-IAEA to EC Commissioner

Euratom advised some Member States in March of continued inspector presence

IAEA

Summary of Consequences for IAEA

Euratom's proposed changes would have a significant impact on:

- Inspections (without Euratom).
- Inspection planning and scheduling.
- Safeguards equipment management.
- IAEA safeguards budget.

Consequences for Inspection Activities

- IAEA would carry out most inspections without Euratom inspectors who would only attend PIVs and a limited number of interim inspections.
- IAEA would need to provide 250 additional person days of inspection to compensate for Euratom's absence.
- IAEA inspectors would need to resolve inspection issues directly with facility operators or State representative, if present.
- Uncertainty exists concerning the smooth implementation of the Additional Protocol, for example Complementary Accesses.

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Consequences for Inspection Planning and Scheduling

- IAEA would have to:
  - Take over the inspection planning in future.
  - Establish better communications with State's safeguards offices.
  - Establish direct communications with facilities in some cases to ensure timely information e.g. about changes in loading of spent fuel into casks for long-term storage.
- IAEA has good experience in scheduling inspections in this way in the EU Accession States.

Consequences for Safeguards Equipment Management

- Euratom would reduce very substantially the use of surveillance, NDA equipment and seals (e.g. at reactors).
- IAEA would need to ensure that effective arrangements for maintenance and supply of equipment are established.
- Surveillance review and seal verification would be performed in Vienna.

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Main Role of Euratom in Future

- Euratom would provide basic SSAC/RSAC functions including:
  - Transmit nuclear material accountancy reports to IAEA (ICRs, MBRs, PILs).
  - Transmit facility design information to IAEA.
  - Transmit Additional Protocol Declarations to IAEA.
  - Audit facility records and item count nuclear material.

IAEA

Resource Implications

- IAEA workload would increase by 5-9 person years (inspectors, technicians).
- $1.5 million/year would be needed for maintenance and replacement of equipment.
- IAEA would need to secure these additional resources, e.g.from Euratom or EU States.

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Presence and Role of Euratom Inspectors at IAEA Inspection?

- Euratom advised IAEA that it will only attend PIVs and a limited number of other inspections.
- Euratom advised some Member States that its inspectors will attend all IAEA inspections (March 2005). IAEA has not been formally notified of this latest revision in Euratom's policy.
- What will be the role of Euratom inspectors in the short- and long-term?

IAEA

Letter from DG-IAEA to EC Commissioner (22 April 2005)

- Changes should be introduced gradually through a due process of consultations.
- High-level consultations needed to reach agreement on principal policy, legal and financial issues.
- Need to ensure that safeguards continue to be implemented effectively and efficiently without negative impact on inspection results.

IAEA

Summary – Euratom Changes

- Euratom's proposed changes would have a significant impact on IAEA and necessary resources need to be secured.
- Need about one year to set up new arrangements with Euratom.
- Need to ensure smooth transition to new cooperative arrangements in consultations between IAEA, Euratom and EU Member States.

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Current Implementation Problems

Inspection Goal Attainment is being negatively impacted by:

- Some communication difficulties related to inspection schedules and activities.
- Delays and reluctances in acceptance of required inspection activities.

There is a need to improve and re-establish sound working procedures and arrangements between all parties.

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Global Status of the Additional Protocol as of April 2005

65 States with AP in force

25 States with APs signed but not yet in force

12 States with APs approved but not yet signed

2 States applying APs pending entry into force

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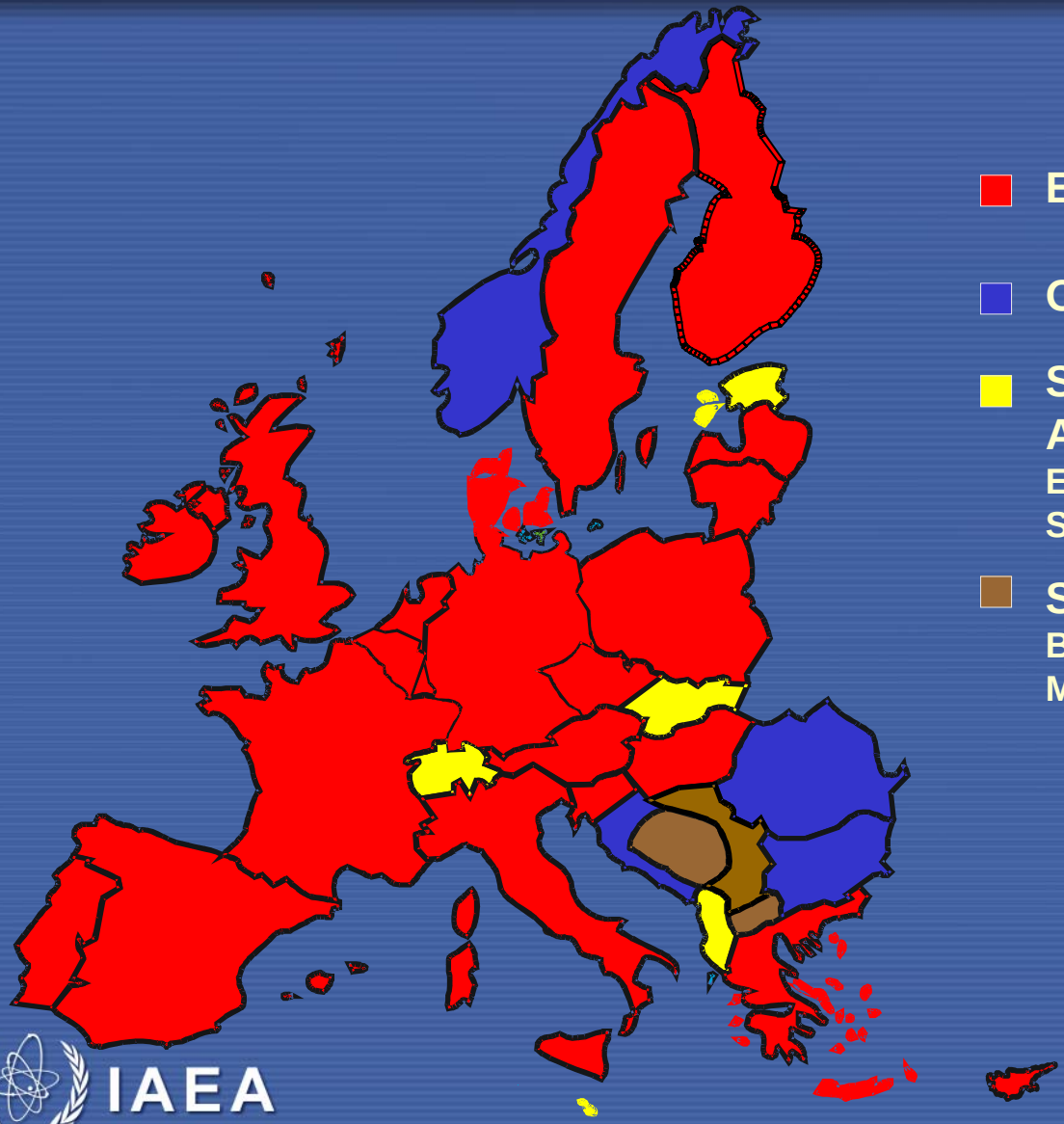
Status of Additional Protocol in Europe as of April 2005



- EU States with AP in force
- Other States with AP in force
- States that have signed the AP but pending entry into force:
  Estonia, Malta,
  Slovakia, (Albania, Switzerland)
- States that have not signed:
  Bosnia Herzegovina,
  Macedonia, Serbia and Montenegro

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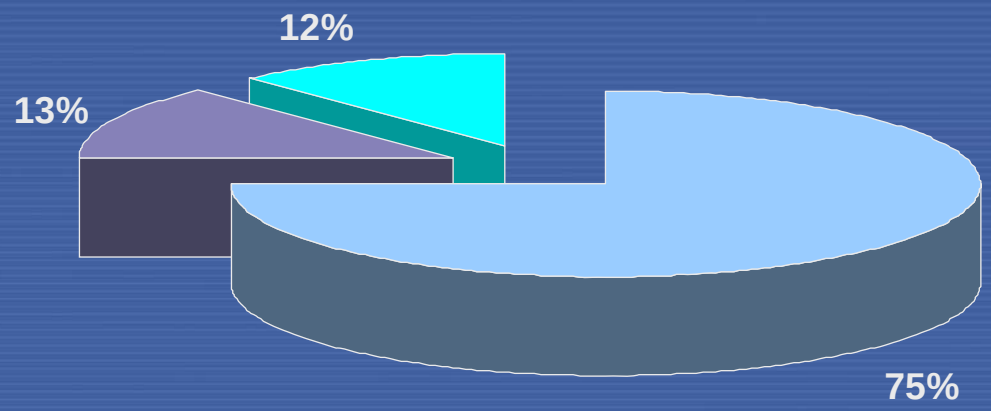
Complementary Accesses (CAs) – Worldwide - 2004

CAs have been carried out without major difficulty in time and scope

124 CAs conducted in 22 States (plus Taiwan) in 2004



[At Sites

At mines and concentration plants and locations with exempted material

At other, e.g. at decommissioned facilities, locations with R&D not involving nuclear material, etc.]

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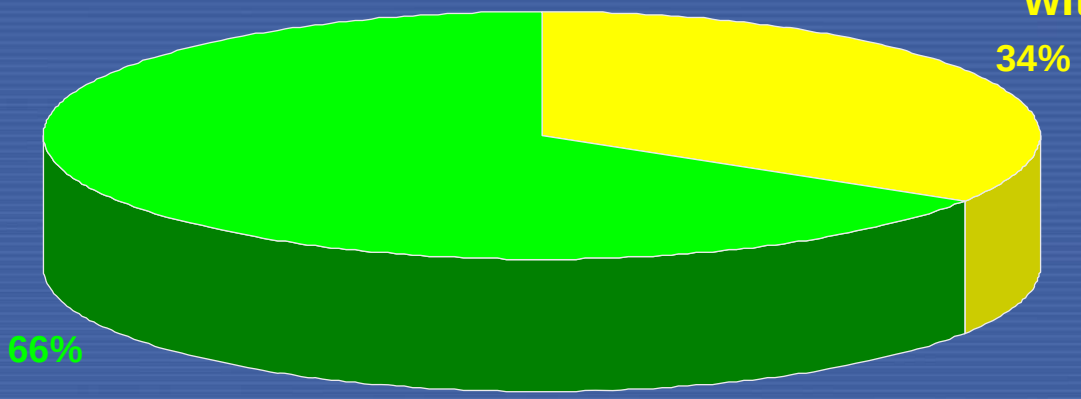
Complementary Accesses – Worldwide – 2004

Carried out at 2 hours notice in accordance with Article 4b(ii) (at other places on sites with nuclear facilities)



Carried out at 24 hours notice in accordance with Article 4b(i) (at all locations including mines and concentration plant, locations with exempted material etc.)

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Additional Protocol Conclusions

- Positive conclusions on the absence of undeclared nuclear material and activities had been drawn for 19 States by end 2003:

  Australia, Bulgaria, Croatia, Ecuador, Ghana, Holy See, Hungary, Indonesia, Japan, Jordan, Latvia, Lithuania, Monaco, New Zealand, Norway, Peru, Poland, Slovenia and Uzbekistan

- It is expected that positive conclusions will be drawn for 2 additional States for 2004.

IAEA

Additional Protocol Implementation in the EU-15 (INFCIRC/193)

- Additional Protocol entered into force on 30 April 2004
- All Article 2 declarations received by 30 November 2004 (due by 27 October)
- Article 2 declarations being reviewed by IAEA
- First complementary access carried out in December 2004

IAEA

Additional Protocol Implementation in EU-15 (INFCIRC/193)

- Declarations generally of good standard.
- IAEA is identifying where improvements or amplifications are needed:
  - Annexed information of all buildings on a site.
  - Issues related to nuclear material exemptions, CAM, and high level nuclear waste (quantity, location).
- IAEA has been consulting with Euratom/ States on these issues.

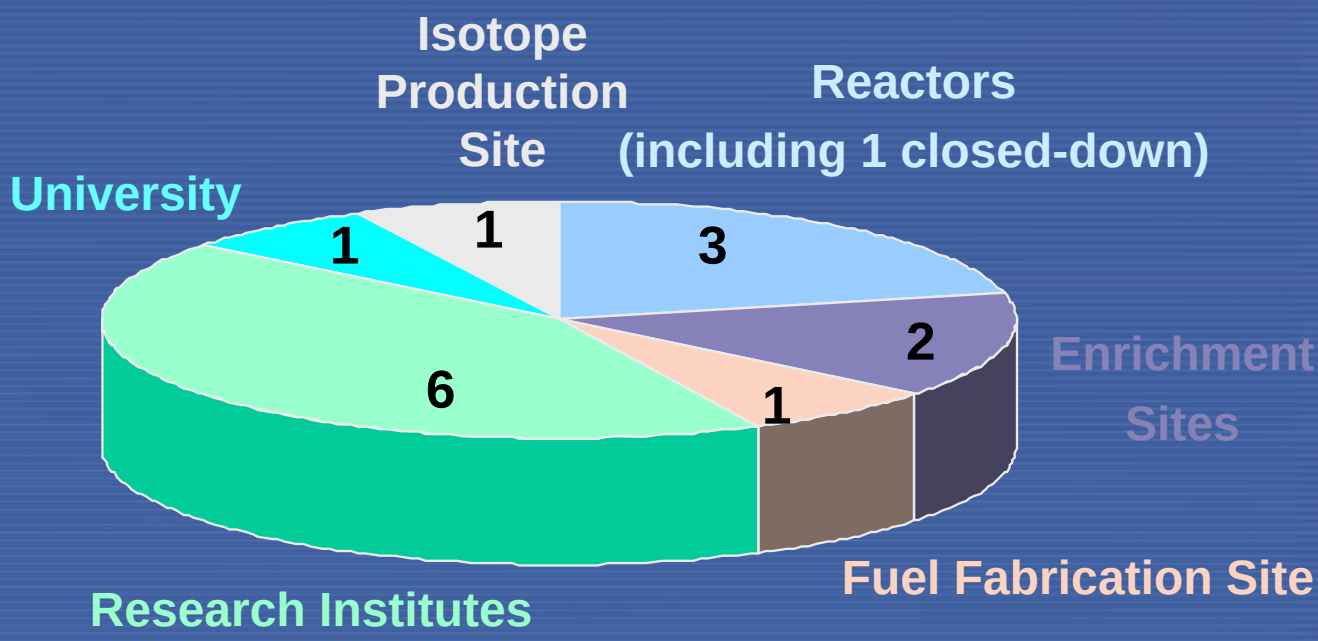
IAEA

CAs performed in EU-15 (INFCIRC/193) as of 5 May 2005

	2 Hour Notice	24 Hour Notice
Austria	0	1
Belgium	2	1
Finland	0	1
Germany	2	2
Italy	0	1
Netherlands	0	1
Spain	1	0
Sweden	1	0
JRC Site	1	0
TOTAL	**7**	**7**

IAEA

Locations of CAs in the EU-15 as of 5 May 2005



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Experience with CAs in the EU-15

- Notification and access arrangements have worked smoothly.
- Activities carried out:
  - Environmental sampling
  - Visual observation supplemented by digital photography
  - Use of hand-held radiation detectors.
- Frequency and coverage of CA's will be expanded in coming months.

IAEA

Safeguards in EU Accession States

Status of Additional Protocol

	AP in force	Conclusion of undeclared nuclear material and activities	Introduction of Integrated Safeguards
Cyprus	✓		
Czech Republic	✓		
Estonia	--		
Hungary	✓	✓	Implemented since Dec. 2004
Latvia	✓	✓	Under preparation
Lithuania	✓	✓	Under preparation
Malta	--		
Poland	✓	✓	Under preparation
Slovenia	✓	✓	Under preparation
Slovakia	--		

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59 CAs between 2001-2005

Safeguards in EU Accession States

Transition period (before INFCIRC/193)

- IAEA is continuing to implement safeguards under existing bilateral safeguards agreements (and Additional Protocols thereto).
- Euratom is attending a limited number of IAEA inspections.

IAEA

Safeguards in EU Accession States

- Transition period expected to continue longer.
- Positive AP conclusions will be drawn, and IS will be established in more States.
- AP and IS implementation will not be affected by accession to INFCIRC/193.
- National safeguards offices will continue to have a role in facilitating safeguards implementation.

IAEA

III. Global Safeguards Issues

- Iran
- Libya
- RoK
- Egypt
- DPRK

IAEA

EVOLUTION OF THE EUROPEAN COMMISSION'S POLICY ON IMPLEMENTING NUCLEAR SAFEGUARDS

C. Cleutinx

European Commission, DG-TREN

INTRODUCTION

For about half a century now, the European Commission has been exercising controls in the nuclear facilities throughout Europe, in accordance with the rights and obligations devolved to it by the provisions of the Euratom Treaty.

This Treaty, which established the European Atomic Energy Community or "Euratom", was signed in Rome on 25 March 1957. It created a system of control mechanisms for nuclear materials that underpinned the rapid expansion of nuclear technologies in the European Community. Chapter VII of the Treaty assigns to the European Commission the responsibility to satisfy itself that nuclear materials (plutonium, uranium and thorium) are not diverted from their intended use as declared by the users.

From the very beginning it has been clear that the Euratom Treaty was not a non-proliferation treaty. First of all it did not forbid the production of nuclear weapons. And secondly, nuclear materials intended for defence were explicitly excluded from verification (Article 84).

Over the years, the nuclear safeguards applied by the Commission's services have nevertheless moved away from purely enforcing the obligations of the Euratom Treaty and followed the path of non-proliferation. The services of the Commission have applied, very explicitly and precisely, the quantitative and mechanistic verification approaches and criteria that have been developed by the International Atomic Energy Agency (IAEA) in the framework of its non-proliferation mandate.

Under the Euratom Treaty, the Commission has to verify that the reports produced by the nuclear operators conform to reality and that they are consistent with the Community's legal provisions. It is a control of conformity. This is distinct, in its principle, to the control of final use, which forms the basis of the IAEA inspections. The Commission has to verify the conformity of an accounting report relative to the physical reality. The IAEA, on the other hand, has to verify that all the nuclear materials present within a non-nuclear-weapon State have been declared in order to make sure that there has been no diversion to clandestine military activities.

There is also another fundamental difference distinguishing them. In the system of IAEA safeguards, the interlocutor is the state. The Agency has to be in a position to draw conclusions regarding the absence of a diversion of nuclear materials at the level of a state.

In the case of "Euratom" safeguards, the interlocutor for the Commission is the nuclear operator. The value of this direct relationship between the control organization and the nuclear operator has been very much underestimated in the past - or at least not all the logical consequences that results from this were drawn. Today it is necessary to place the nuclear operator where he belongs and to emphasize his role as the first line of defence against the diversion of nuclear material.

The existence of the European Union, with the largest single internal market in the world, has also added significantly to the Community's safeguards by facilitating for example the creation of "Multilateral Nuclear Approach joint facilities" or MNAs, which cross the borders of the EU Member States. The "peer control" built into such MNAs (e.g. Urenco, Eurodif) are solid examples of a strong contribution brought by EU companies to nuclear safeguards.

In the future, the entry into operation of the European Satellite Navigation System “Galileo” will also add significantly to EU nuclear security.

But there is still much more to “Euratom” nuclear safeguards.

The Commission's nuclear control activities fully apply to all civil nuclear installations, whether they are located in nuclear-weapon States or non-nuclear-weapon States in the EU.

By pursuing as it did in the past this non-discriminatory control in nuclear-weapon States, as well as in non-nuclear-weapon States, the European Union will continue to set an unprecedented example in the world by ensuring a most global and robust system of supranational nuclear safeguards.

One can also safely say that today the EU is the most “nuclear inspected” area in the world. In 2004, some 7 000 person-days of inspection were performed in EU nuclear installations by the European Commission and/or the IAEA. This is about the same number as were performed by the IAEA in the rest of the world!

On top of this, in a majority of EU Member States, dedicated national authorities are also carrying out controls on the nuclear operators which are similar to those performed by the Commission.

Also, the number of Commission staff involved in nuclear safeguards has been increasing over time. There are now about 190 nuclear inspectors, compared to 160 in 1990, and this will increase further with the recruitment of additional inspectors from the new Member States.

And finally, all the services of the Directorate General for Energy and Transport that are involved in nuclear matters have now been brought together in Luxemburg. Concentrating more than 300 highly qualified officials together, mutually reinforces the different activities in the nuclear field, such as safeguards, radioprotection, and safety, and is encouraging the development of important synergies.

It is against this background that the European Commission still intends to improve, both in terms of efficiency and effectiveness, its contribution to nuclear security.

THREE DIVERSION SCENARIOS

Bearing in mind the need to refocus back onto its own responsibilities, the Commission initiated an internal review in September 2000 of the implementation of its rights and obligations under Chapter 7 of the Euratom Treaty.

As a result of this review, the Commission adopted, in April 2004, the principles of a safeguards approach that is better focused on the objectives of the Euratom Treaty.

It is important to stress that this is the first time since 1958 that the Commission has taken such a firm position. This will more clearly define and reinforce the important responsibilities it has in the framework of nuclear safeguards.

A number of scenarios for the diversion of nuclear material from declared use have been identified in order to define the extent and the nature of the Commission’s role and responsibilities in relation to them.

In the first scenario, an individual or subgroup within or outside the operator’s organisation diverts (steals) nuclear material and/or introduces falsified accounting data, or makes mistakes. The prime responsibility for the prevention (ex ante) or detection (ex post) of such a scenario is with the nuclear operator himself acting in conjunction with the relevant State authorities. In order to cover this scenario, the operators must have both physical protection

measures in place, and nuclear material accounting and control systems which function to the required standards and which will benefit from the Commission's new audit activities.

In the second scenario, the operator, as an organisation, diverts nuclear material for purposes other than the declared use. Such a diversion from the declared use, although improbable, always remains a possibility. Under the Euratom Treaty, detection here is the responsibility of the Commission in co-operation with the Member States. The Commission services continually verify, through auditing, that the nuclear operators implement nuclear material accounting and control systems which are able to reflect, at any time, the physical situation of the nuclear material. In order to satisfy itself about the non-diversion of nuclear material, the Commission services send inspectors on-site to verify the measurement and recording procedures that support the declarations on nuclear material made by the operators of nuclear installations. These inspectors also perform frequent physical verifications to reach conclusions independent of these procedures. The Commission services also use any other information that is available, be it from open sources or input received from national entities on the occasion of their own verification activities or from international organisations, mainly the IAEA.

In the third scenario, the Member State diverts nuclear material for non-peaceful use. This scenario would constitute a breach of the commitments of the Member State under Article 3 of the Non Proliferation Treaty (NPT) for the non-nuclear-weapon States. The detection of such a diversion is the formal responsibility of the IAEA. Under this scenario, the Commission acts as the collective "State system of accounting for and control of nuclear material" (SSAC) for all EU Member States. The relationship with the IAEA under this scenario is covered by the three safeguards agreements, which also take into consideration the voluntary offer for the EU nuclear-weapon States. One has to add, however, that the non-discriminatory controls performed independently by the European Commission in the civil nuclear fuel cycles of the two EU nuclear- weapon States are an important additional guarantee to EU nuclear security.

TOWARDS AN OPTIMAL PROTECTION OF THE EUROPEAN CITIZEN

The primary objective of a credible nuclear security system will always remain the protection of the European citizen and the respect of EU responsibilities towards the international community. To accomplish this, it is important to bear in mind that nuclear security is a concept broader than nuclear safeguards.

Nuclear security in the European Union therefore has to be considered as a partnership as the responsibilities are spread over several different actors:

- Firstly, chapter VII of the Euratom Treaty dealing with safeguards only covers some materials, namely plutonium, uranium and thorium.

- Secondly, the Commission currently has no responsibilities concerning the physical protection of nuclear material, such as measures to protect against nuclear terrorism. This matter remains the sole responsibility of each Member State.

- Thirdly, non-proliferation control covers entire States under the Non Proliferation Treaty and is under the mandate of the IAEA. But it is clear that the excellent work performed by the European Commission in the EU through the unique expertise and experience of its nuclear inspectors and the responsibility of the Commission as a collective SSAC bring important contributions to the IAEA mandate.

- And finally, Commission inspectors are not in charge of the quality control of nuclear material management systems at nuclear installation. This remains the principal responsibility of the nuclear operators.

Therefore, only a partnership between the various actors in which the responsibilities, where they belong and how to co-ordinate them, are very clearly defined, provides the best guarantee to the European citizen.

EURATOM SAFEGUARDS: A DYNAMIC APPROACH

Nuclear safeguards inspections have, in the past, clearly included responsibilities that are beyond the scope of the Euratom Treaty. This growth in the activities covered, at the technical level, by highly qualified and motivated Commission inspectors has however carried the danger that responsibilities become unfocused.

The implementation of a dynamic safeguards approach will result in an evolution from a system based purely on quantitative objectives to a more qualitative system that is flexible and capable of adapting to evolving circumstances. The application, by the Commission, of audit tools and methods in the evaluation of the operator's nuclear material management system provides an added value to the assurances the Commission is required to obtain about the proper use of nuclear material. Similarly, by tackling quality problems at their root, the resulting implementation of safeguards is improved. This is equally beneficial for the IAEA. It is interesting to note that such a change also fits well into the development of international safeguards arising from the implementation of the additional protocols to the IAEA safeguard agreements. The traditional systems of safeguards based only on quantitative objectives clearly demonstrated their weaknesses as they have for example considered the operator's internal systems too frequently simply as a black box.

EU nuclear safeguards must offer a more effective and transparent mechanism. It must make it possible to modulate the degree of intrusion depending on the performance of the nuclear operators and on the experience that has been gained by the Commission services.

This scheme will take into account the potential risk presented by the nuclear operator, rather than only the technical features of an installation or the sensitivity or the quantity of the nuclear materials that are treated or stored.

It is also based on concrete elements specific to each nuclear operator and no longer only on theoretical schemes worked out using mathematical models.

In this framework particular attention will be given to installations holding small quantities of material. Experience has shown that quality related problems arise more frequently in these installations that often do not have a robust nuclear management system.

This situation calls for a renewed screening of these installations and an upgrading of the related effort, on a case by case basis. Under current geopolitical circumstances tighter control on “locations outside facilities” must be an utmost priority even if intellectually less fascinating - though more demanding for the safeguards professionals - than “routine” controls in large nuclear installations. Hence the new approaches will also reallocate additional resources to this type of activity.

AUDIT METHODS AND TOOLS

The European Commission's new approach to nuclear safeguards combines an audit of the accounting documents that are produced from the nuclear operator’s system of accountancy and control with random checks of the procedures and direct physical verifications, the latter remaining an essential element of the control scheme. By reinforcing the nuclear operator as the first line of responsibility for the control within his facility, the Commission services can intervene at a second level to draw conclusions on the practical implementation and effectiveness of the operator’s procedures.

The audit activities will fully exploit all data in the possession of Commission services as well as their knowledge of the facilities. The Commission services have the longest experience at an international level in safeguarding nuclear material. The use of this “experience data" provides important information on the current management of the nuclear materials, particularly with respect to its reliability. The historical performance of a nuclear operator can and must be an important indicator for the evaluation of his current system.

RATIONALIZED INSPECTION EFFORT

An effective rationalisation of the inspection effort needs to have recourse to unannounced inspections or to short notice and random inspections.

The Commission can also make use of the operator’s own system of measuring and sampling to draw its conclusions. The inspection activities will not be limited to physical access to the nuclear materials but will also comprise specific audit activities. If the results of the verifications show deficiencies in the operator’s system of management or reveal discrepancies with the information declared, the extent and the intensity of the verifications will be increased. The Commission services can also take into account the results of other control organizations, when these are available, in order to confirm their own judgment regarding the absence of a diversion of materials.

Nuclear control activities can lead to requests for explanations or the implementation of corrective actions on the operator by the Commission services. These requests will be subject to a rigorous follow-up. The absence of a satisfactory reply or of the implementation of the corrective actions required by the Commission can lead to the use of the provisions provided for in Article 83 of the Treaty. The operator who fails to comply will be sanctioned. If the situation does not improve, the Commission, pursuant to Article 82 of the Treaty, will then ask the Member State, and not the operator, to take the necessary steps to stop the infringement. As a last resort, the Commission reserves the possibility of referring the matter to the Court of Justice. The existence of these coercive powers within the framework of the Euratom Treaty is one of the unique features of European safeguards and has an important dissuasive effect.

CO-OPERATION WITH THE IAEA

The inspection activities of the IAEA are co-ordinated with those of the Commission services, within the limits of their mutual responsibilities and in the framework of the trilateral agreements. If the IAEA considers it necessary, in order to fulfil its own objectives, to intensify its inspection effort beyond what is useful for the Commission under Article 77 of the Treaty, specific arrangements can be set up. It would, in particular, be possible to ask the operator or the national authorities to be present during these additional verifications of the IAEA, with an obligation to inform the Commission on the outcome of such inspections. These arrangements would have to be implemented on a case by case basis.

In fulfilling their mandate, the IAEA shall, according to Article 3 of the safeguards agreement "… take due account of the effectiveness of the Community's system of safeguards…". The IAEA is, however, not restricted to only "observing" the Commission's inspection activities. Article 14b of the Protocol to the safeguards agreements allows the IAEA to go beyond simple observation and in particular to perform additional activities or inspections, provided that these are agreed beforehand with the Commission and the EU Member States party to the Agreement.

However, Article 81 of these agreements also foresees the possibility for the IAEA to make use of "state specific factors" or what we can call EU specific factors to adjust “the actual number, intensity, duration, timing and mode of routine inspection”. This is what can be defined as differentiation within a non discriminatory system.

Under the terms of the new approaches to nuclear safeguards, the Commission's services will continue, as in the past, to co-operate effectively and efficiently in inspection and technical matters with the IAEA. This cooperation will be even more fruitful than in the past as a result of a more complete mix of inspection approaches and thus less redundancies between IAEA and European Commission inspections.

CONCLUSION

The evolution in the implementation of nuclear safeguards in the framework of the Euratom Treaty is based on the success of the work undertaken by the Community since the signing of this Treaty. It should also take into account the tremendous added value to nuclear security brought by the European integration process.

Commission safeguards today have to focus on the mission that was entrusted to them by the founding fathers of the Treaty. This development, which testifies to the capacities of adaptation of the Euratom Treaty, will strengthen its credibility and effectiveness and will guarantee to the European citizen and to the international community the highest contribution of EU nuclear safeguards to nuclear security.

Session 2

Safeguards Systems - Past, Present and Future

Implementation of the new Euratom Regulation No 302/2005: The challenges ahead

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Abstract

The new Euratom Safeguards Regulation was published in the Official Journal L54 on 28.02.05 under the number 302/2005.

With the publication, a lengthy procedure involving initial drafting by the Commission, discussions with the stakeholders, approval by the European Council and final approval by the Commission was completed.

The new Regulation was introduced as a means of adapting to the new legal framework, modern techniques, and controls commensurate to the importance of the nuclear material.

The implementation of the new Regulation imposes a number of challenges to both the Commission and the operators:

The Commission has to play an integrative role, in the direction of having the accountancy and reporting systems of the operators adapted to the new Regulation. The different starting point of the EU15 operators to the EU10 has to be taken into account. The appropriate support projects (ACCESS for EU10 and ENMAS for EU15) have been launched.

The database at Headquarters has been adapted to accept reports in both 3227/76 and 302/05 format, and will continue to operate in this mode, until the end of the transitional period of 3+2 years.

The new Regulation and the Guidelines for its implementation will allow the operators fulfil the legal requirements stemming from the Additional Protocol, especially those related to the site declaration and the waste transactions.

The provisions of the Regulation on derogations and waste declarations will allow to adapt the reporting requirements to the strategic value of the material, while taking into account the concerns of the international community, as expressed with the IAEA policies, especially integrated safeguards.

Altogether it is expected that with the efficient implementation of the new Regulation, the EU member states and the EU as a whole will keep their high non-proliferation credentials.

Keywords: Euratom; Safeguards; Regulation; Reporting; Waste; Derogation.

1. Introduction

The new Euratom Safeguards Regulation was published in the Official Journal L54 on 28.02.05 under the number 302/2005.

It is recalled that according to Article 79 of the Euratom Treaty "The Commission shall require that operating records be kept and produced in order to permit accounting for ores, source materials and special fissile materials".

The safeguards Agreements[1,2,3] between Euratom, its Member States and the IAEA stipulate that the Community shall provide the Agency with information concerning Nuclear Material, while the Protocols Additional to the Agreements[1a, 2a, 3a] put the requirement for the provision of declarations related to the nuclear material to the Member States and the Community.

Finally, the agreements between Euratom and third states (e.g. USA, Canada, Australia) do also stipulate the provision of information related to nuclear material from Euratom to the Third states.

The new Euratom Safeguards Regulation is thus the tool which will allow the operators to comply at the same time with the requirements of the Euratom Treaty of the Safeguards agreements and their APs, while it gives to the Commission the legal instrument to discharge its commitments vis-à-vis its international agreements.

2. The changes introduced and their timetable

The new Regulation has to be applied from its date of entry into force by all 25 Member States, with the exemption of the reporting formats of ICRs, PILs and MBRs, for which a transitional period of up to 5 years is given to the operators of the "old" Member States.

Concerning BTCs, the annexes of the new Regulation are basically the same as those of the old one, apart for the numbering and the declaration of "Use". Thus, no update of the BTCs would be required until the new reporting format would be adopted by our operator.

An exemption to this are the BTCs of mines and waste installations (new), which would have to be submitted within 120 days after the entry into force.

Concerning PSPs, existing ones remain in force, while the new ones agreed will have to follow the new Regulation. Existing PSPs will have to be adapted to the new Regulation at the latest when the new reporting format and the new IC codes are adopted by an operator.

Concerning the new Waste Policy (BTCs, records and reports), an initial stock list of waste is required within 120 days of the entry into force, while, not registered waste treatment and storage installations will have to submit their BTC and initial stock list within the same dead line.

Concerning Additional Protocol related reports, the site declaration and their annual updates should follow the new Regulation, while for conditional waste an initial stock is required, to be followed by the first declaration of change of location of conditioned waste for the period to the end of 2005 (to be submitted in 2006).

Finally, the new derogation mechanism applies, and operators who qualify for a derogation could, at any time, send their request.

3. Waste reporting

The aim of the waste reporting scheme introduced in the new Regulation is twofold: meet the obligations under the Euratom Tready and the Additional Protocols but also reduce the reporting requirements to a level commensurate with the strategic value of the material.

Article 2 of the new Regulation provides clear definitions of waste (Retained, Conditioned, discards to the environment) on the basis of its conditioning, recoverability and suitability for nuclear use.

Article 30.1 introduces the obligation for the initial stock list declaration, while Article 30.2

provides to the waste handling and storing facilities a derogation from providing accountancy reports and describes the simplified records that have to be kept at the facilities.

Article 32(a) and 32(b) describe the annual reports of receipts and shipments of conditioned waste required, while Art. 31 and 32(c) are related to reports of waste processing and changes of location required by the Additional Protocol.

For retained waste the reporting requirements and inspections, can be summarised as follows:

- TW upon transfer to retained waste inventory
- Conduct processing of waste (no separation of elements) outside the main inventory
- FW upon return to main inventory, followed by SD or SF
- RD or RF upon receipt, followed by TW
- Initial stock list by category, storage area and type of waste
- Annual PIT-no re-measurements
- Program of activities
- BTC verification
- Normally, no physical verifications, except to resolve discrepancies.

For conditioned waste the requirements are:

- TC upon transfer to conditioned waste inventory
- Where applicable, IAEA safeguards terminated
- FC upon return to main inventory
- Annual reports of shipments/receipts (Annex XIII, XIV)
- Initial stock list by category, storage area and type of waste
- Annual PIT-no re-measurements
- Program of activities
- BTC verification
- Normally, no physical verifications, except to resolve discrepancies.

Further, the new Regulation provides the possibility of termination of Safeguards for waste containing very low concentration of Nuclear Material.

4. Derogations from the form and frequency of reporting

The Commission may grant operators a written derogation from the rules governing the form and frequency of notifications (ICRs, MBRs, PILs etc). The derogation is granted for a Material Balance Area holding:

- Small quantities of NM
- D,N,T in non-nuclear activities
- Special fissile material in Gram quantities
- Pu-238>80% isotopic concentration.

The operators having obtained a derogation are instead required to submit a yearly report using Annex X and also to request the derogation of new articles that are permanently added to their stocks.

The Commission is charged with further requesting and obtaining exemption of the derogated MBAs from IAEA Safeguards and the reporting of exempted material under the AP to the IAEA.

5. The new Reporting requirements

The new Regulation introduces a number of new requirements on ICR, MBR and PIL reporting. In particular:

- An ICR with the BA split per obligation is required every month. The No Change declaration is discontinued
- The MUF resulting from a PIT must be reported in the ICR following the PIT (NM practice is discontinued)
- Two ICRs are required if the PIT is not the last day of a month
- There should be at least one PIT per calendar year, with a maximum of 14 months between PITs
- The MBR has to be drawn up per obligation

- The Gram should be used as the only weight unit. The implementation of this requirement could wait until the introduction of the new reporting format
- Electronic reporting is mandatory.

Concerning reporting using the new format, Art. 39 of the Regulation provides for a transitional period of initially 3, but extendable to 5 years. During that period the formats of the old Regulation could be used.

Nevertheless, during this transitional period, operators are allowed to use the IC codes of the new Regulation with the old format of reports. This will be particularly beneficial towards the implementation of the new waste policy.

It should be mentioned that the Euratom Headquarters data base has been adapted so that it can receive and treat reports under the old and new formats, while it is maintaining a link to the "legacy" declarations and provides the possibility of correcting under the new Regulation accountancy lines that were reported under the old Regulation.

The Commission has decided to provide assistance to the operators of Nuclear installations towards meeting the requirements of the new reporting format.

In particular for the operators of the 10 new Member States, the project ACCESS providing appropriate hardware and software is reaching completion.
ACCESS offers the following functionalities:

- Local NMA database (Oracle)
- Data input via Editor or Xloader
- Local Rulebook validation
- Automatic calculation of MBR and stocks
- Electronic reporting to Euratom/rapid feedback
- Secure data transfer (encryption)
- Automatic update of Rulebook and software.

For the operators in the old Member States the Commission has initiated the project ENMAS, mainly addressed to small and medium sized operators.
For **small operators**, ENMAS is viewed as a stand alone application, with manual NMA editor, validation of the good syntax of the declarations, some type of Rulebook, XML report generation and viewing. Encryption and sending by email can be independent.
Alternatively, ENMAS could be a Web based application, with appropriate security keys.
For **medium sized operators**, who currently produce the accountancy reports as a by-product of their NM management system:

- the X-loader solution, for operators who do not wish to change their NM management system (pitfall: some data fields are missing – e.g. CRC)
- the integration of an ENMAS routine in the system (for operators who will use the opportunity of the new Reporting format, to develop a new NM management system).

For **bigger operators**, ENMAS will be the forum for technical discussions, and definition of the requirements. They should be able to develop themselves what is needed.

It can thus be expected that by the end of the transitional period all operators will be reporting using XML labelled format. The Rulebook validated reports with intrinsic QA features will definitely reduce the need for corrections and Commission interventions.

6. The Guidelines

As foreseen in Art. 37 of the Regulation, the Commission shall adopt and publish Guidelines for the application of the Regulation. The text of the Guidelines, as agreed with the stakeholders during numerous discussions before the Regulation was approved, will be published as a Commission Recommendation, in all Community languages. It is expected that this procedure will be completed before the end of 2005.

7. Conclusions

Full implementation of Regulation 302/2005 in the EU, will assure compliance of the operators towards the Euratom Treaty and the international agreements, but also compliance with the obligations of the Community towards its international safeguards obligations.

References

[1] Agreement between the 13 Non-Nuclear -Weapon State members of the EU, Euratom and the IAEA in implementation of the Treaty of Non-Proliferation of Nuclear Weapons, published in OJ L51, 22.2.78

[1a] Protocol Additional to above Agreement[1] published in the Official Journal no. L67/1 of 13.3.99

[2] Agreement between the United Kingdom, Euratom and the IAEA in connection with the Treaty of Non-proliferation of Nuclear Weapons, published as INFCIRC/263

[2a] Protocol Additional to above Agreement[2] , published as COV/1998/30, 29.5.98

[3] Agreement between France, Euratom and the IAEA for the application of safeguards in France, published as INFCIRC/290.

[3a] Protocol Additional to above Agreement[3], published as GOV/1998/29, 29.5.98

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Performance-Based International Safeguards System[1]

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Abstract

Many have criticized the current system of international safeguards as being too prescriptive, particularly in the context of integrated safeguards (IS). As safeguards are transformed, performance criteria have been proposed as an alternative, but to the extent that implementation options have been discussed previously, they have been limited in detail. Adoption of a performance-based system could, in theory, produce significant benefits. Flexibility in meeting performance criteria could enable certain efficiencies in safeguards application. Moreover, clearly linking safeguards actions to detection objectives could make the application of inspections more effective. By specifying the purpose of safeguards activities, implementation issues and follow-up actions should become clearer. Such an approach could also have a significant impact on reporting. A "verification goal" could be expressed on a material basis, e.g., in terms of the fraction of material to be directly verified with some frequency. A "deterrence goal" could be expressed in terms of retaining a meaningful detection capability across credible acquisition paths (and concealment scenarios) that would be specified in the criteria. Illustrative examples of performance criteria are given and the implications of a performance-based system are discussed in the paper.

Introduction

The current international safeguards regime is in transformation, with new authorities and tools provided under the Additional Protocol and the International Atomic Energy Agency's (IAEA's) efforts to develop integrated safeguards (IS). Safeguards have been criticized for being too prescriptive, particularly in the IS context. The safeguards criteria describe the set of activities to be performed during inspections. Inspectors perform the prescribed tasks and report their findings. These findings are summarized annually in terms of Agency success in meeting certain inspection goals. It has been argued that this approach results in a mechanistic system more focused on "checking off" inspector activities from a prescribed list than genuinely safeguarding the nuclear fuel cycle against proliferation.

As the system changes, the prescriptive approach is increasingly inadequate. As the IAEA moves away from the worst features of such an approach, there is no guiding principle for future safeguards. In this context, the adoption of performance objectives has been proposed as an alternative means for organizing the safeguards system. There are now real opportunities for a debate on this approach to safeguards. A performance-based system would clearly state the technical objectives of the safeguards system and describe a means for assessing whether or not those objectives have been fulfilled. The safeguards measures used to meet those objectives would not be fixed. Different measures could be used in a dynamic manner to meet the same objectives in different states.

Given the dual safeguards objectives of verification and deterrence, it is proposed that performance objectives be expressed in terms of (1) nuclear material inventoried; and (2) proliferation activities covered. A "verification goal" could be expressed on a material basis, e.g., in terms of the fraction of material to be directly verified with some frequency. A "deterrence goal" could be expressed in terms of retaining a meaningful detection capability across credible acquisition paths (and concealment scenarios) that would be specified in the criteria. This would deviate from the current focus on quantity and timeliness goals. It is argued that these goals as currently formulated produce unnecessary confusion in safeguards implementation and reporting. Furthermore, they do not indicate the overall performance of the safeguards system, particularly in terms of deterrence.

The Objectives of Safeguards

As performance is directly related to the objectives of safeguards, it is useful to recall that, in practical terms, the function of international safeguards has been described as fulfilling two primary roles for Member States: (1) to verify nuclear activities in a State to provide independent certification of that State's nonproliferation bona fides; and (2) to deter States from proliferating via a risk of early detection.[2]

In many ways these objectives are two sides of the same coin. The activities that the Agency performs are intended to encompass both functions. However, they stem from very different motivations. The first refers to a State's own desire to dispel suspicion about its activities.[3] The second stems from a desire to discourage states from proliferating. Depending on the State in question, either objective may ultimately determine the requisite safeguards for a State.

Safeguards Criteria and the Safeguards Implementation Report (SIR)

Currently, the safeguards criteria describe the set of activities to be performed at the declared facilities in a State. These activities are determined directly by the facility type as well as material type and quantity. When fully implemented, these safeguards measures by definition meet the requirements of safeguards.

On an annual basis, the SIR reports on what was actually done. Based on the performed activities, the Agency determines "goal" attainment. Overall inspection goals are met if all criteria are satisfied (i.e., planned safeguards activities are implemented) and all anomalies are resolved in a timely manner. While certainly a State's cooperation influences safeguards implementation, current reporting focuses more on the Agency's performance.

Critique of the Current System

The combination of prescriptive safeguards criteria and reporting that focuses on quantity and timeliness goal attainment (as currently defined) has a number of shortcomings.

Overall safeguards system design/performance is difficult to assess.
The current criteria/SIR system focuses attention on Agency activities rather than overall safeguards system effectiveness in meeting safeguards objectives. The criteria specify what is to be done but not why. The current reporting linkage is between safeguards activities and inspection goal attainment, not between safeguards activities and the mitigation of proliferation concerns.

This disconnect presents a number of problems. Quantity goal attainment has evolved as the most important measure of performance. As noted above, attainment of this goal relies, in the first instance on execution of prescribed criteria activities. The criteria define the activities for goal attainment and goal attainment is achieved by meeting the criteria. Goal attainment becomes the end in and of itself rather meeting any functional requirements.

Linkages between goal attainment and safeguards objectives are unclear
Under the current system, the significance of inspection goal attainment—or failure—is not clear. The relationship between these inspection goals and the safeguards objectives of verification and deterrence is left unspecified.

It is not possible to gauge the sufficiency of the system design measured against the objectives of safeguards. What level of confidence should be associated with safeguards conclusions if goals are attained? How much material has been directly verified—100 percent? 50 percent? How strong a deterrent has been provided? Has deterrence been provided at the 90 percent level? The 20 percent level? How can we ascertain deterrence? What is acceptable?

Deterrence issues are not adequately captured in the SIR
It is often assumed that timely detection (and thereby deterrence) is achieved if the timeliness goal is attained. However, as currently expressed, timeliness goal attainment simply means planned inspections

were performed on schedule and anomalies resolved swiftly. The degree to which the inspection plan actually provides timely detection of various proliferation actions is not addressed.

To a degree, from a deterrence perspective, whether planned activities/inspections were fully implemented or not may not be critical. It has been pointed out that if the Agency simply chose not to perform a particular scheduled interim inspection, deterrence objectives may still have been met, even though the timeliness goal was not attained. In addition, random inspections can, in principle, provide detection opportunities for certain scenarios that scheduled inspections cannot. However, it must be noted that if the actual frequency of inspection, scheduled or random, is observed by the proliferator to be quite low, deterrence will not be credible.

The current approach provides no direction for State-wide safeguards design and implementation
With the adoption of the Additional Protocol, the scope of safeguards has been unambiguously extended to include the detection of undeclared facilities. Therefore, there is a need to extend international safeguards approaches to meet this task. This is clearly a work in progress at present, but a prescriptive approach is particularly ill suited to this task.

Performance-Based Safeguards Objectives

Performance objectives offer an alternative to prescriptive means of meeting organizational goals. Rather than specifying in detail the exact set of activities that de facto represent goal attainment, care is taken in specifying the goals and multiple means for meeting the goals are allowed. The emphasis is placed on meeting functional requirements rather than the means by which they are attained—on results rather than process.

What would a performance-based system look like?
To be useful, organizational goals must be translated into technical parameters against which the effectiveness of a set of actions can be designed to and measured against. These parameters must capture the true functional requirements of the system. For international safeguards, it is proposed that separate performance goals be articulated for the verification and deterrence objectives of the system.

In terms of verification, the safeguards system should periodically demonstrate that virtually all declared nuclear material in a State are accounted for in the declared nuclear program. Materials accounting would, of course, be the primary tool utilized for this function.

In terms of deterrence, the safeguards system should provide adequate assurance that credible proliferation actions, should they occur, would risk detection in a timely manner. For deterrence, possible future actions are the focus of safeguards and not historical nuclear material inventories. The scope of actions to be detected (including concealment strategies) should be broad. Certainly it would include diversion and misuse at declared facilities as well as undeclared, off-site activities. Using pathway analysis for this purpose, where paths represent the set of activities a State must undertake to produce weapon-usable material, is recommended. The activities to be detected would be informed by facility assessments as well as resources such as the physical model. Table 1 summarizes the differences between these two performance objectives.

	Safeguards objective	Scope	Role of "quantity" goal	Role of "timeliness" goal	Primary safeguards measures
Verification	Verify that all declared material remains in civilian use	Declared materials	Determines measurement level – establishes target values	Not particularly important; verify inventory with some frequency	NMA augmented by C/S (for efficiency)
Deterrence (through risk of timely detection)	Establish the ability to detect all credible proliferation pathways	Undeclared actions at both declared and undeclared sites	Not a focus per se, only to baseline the paths in terms of production rate (e.g., 1 SQ/yr)	A major focus; objective is to detect pathway use prior to path completion	Surveillance/ NMA/ unannounced inspections/ new measures

Table 1. Comparing Verification and Deterrence Goals

Establishing separate goals would produce several benefits. A direct linkage would be produced between individual inspection activities and safeguards objectives. This would simplify the application of safeguards and enable more effective and flexible implementation. The level of deterrence afforded by the system would also be made clearer. This would have the effect of raising the visibility of this important, but poorly characterized, safeguards objective.

Although verification and deterrence are separate goals, they are mutually reinforcing. For some proscribed activities, it may be the case that verifying material is the most effective way of covering a proliferation pathway. Conversely, an accurate inventory may rely heavily on assurances that certain concealment activities have not occurred.

How would performance goals be expressed? At the State level? At the facility level?
Under integrated safeguards, there is a desire to develop State-level safeguards approaches. Generically, at the State level, the performance goals could be articulated in the following manner:

For verification:

1. All nuclear material will be accounted for at a specified level of accuracy (e.g., International Target Values).
2. Inventory verification will take place periodically, not to exceed X years.

For deterrence purposes:

1. All credible acquisition paths (and concealment scenarios) in a State will be covered by safeguards measures.
2. Detection probabilities prior to pathway completion for these paths will meet Y level.

Although the description above refers to all material and all paths, different goals can be established for different materials and paths. As in current safeguards, a requirement for greater confidence in safeguarding un-irradiated, direct-use inventories could be expected. Similarly, deterrence from using some paths may be judged to be more important than others.

Verification Goals
In establishing specific verification goals, two parameters must be fixed: the level of verification and the frequency of inventory taking. The verification level could be set consistent with best available measurement practice (e.g., International Target Values). The parameters could also be selected consistent with the current system with higher verification levels required for more sensitive material.

Frequency of inventories could be set according to member state wishes. The current physical inventory verification requirements suggest an annual update of inventories. This could be modified accordingly for different material types. Table 2 describes possible State-level verification objectives.

Material Category	Verification Level	Verification Frequency
Unirradiated Direct Use (UDU)	High	Biannual
Irradiated Direct Use (IDU)	High	Annual
Indirect Use	Medium	Semiannual

Table 2. Illustrative State-Level Verification Objectives

Facility-specific inspection requirements for verification purposes would follow from the State-level goal. Of course, for dynamic inventories, more frequent inspections may be necessary to maintain accurate accounting.

Deterrence Goals
In establishing specific deterrence goals, it is necessary to establish two different parameters: the proliferation actions to be deterred and the desired timely detection probability.

This system can be thought of in the following manner. The proliferator is looking to make relatively long-term, expensive decisions in terms of weapon program design. He must choose the type of material desired, the strategy for production, the means for acquiring feed material, etc. The Agency would like to ensure that the probability of timely detection is sufficiently high that a decision to proliferate is not taken. Failing that, it would appear to be more desirable that pathways involving declared material and facilities are not utilized.

The actions would specify what the safeguards system is designed to "see." Defining the relevant proliferation actions is a challenging task. Nonetheless, there are clear ways to proceed. Expressing pathway detection goals at the State-wide level is natural. As pathways represent the set of activities a State must perform to produce weapon-usable material, proliferation pathways themselves must be defined in State-wide terms.

Not all paths are equal in terms of their attractiveness. As with verification, where goals were established on the basis of material types, detection requirements can be set for different "classes" of paths. Uniformity can be achieved by setting the same detection requirements for paths of the similar difficulty in execution.

Pathways involving…	Relative Technical Difficulty	Timely Detection (Deterrence) Requirement
Clandestine reactor and/or enrichment facility	High	Low
Clandestine reprocessing facility	Medium	Medium
Only clandestine conversion facilities	Low	High

Table 3. Illustrative State-Level Deterrence Objectives

Table 3 describes possible State-level deterrence objectives. Timely detection here is taken to be the probability of detection prior to pathway completion. This, in effect, defines a required level of deterrence for each path. Again, the logic is that easier paths must be deterred to a higher degree. For a safeguards system to be acceptable, it must be demonstrated that the package of safeguards measures implemented in a State will meet these requirements. The identified paths of concern must be covered at the desired level (or better) in order to provide the required deterrence.[4]

It is best to think in terms of requirements for pathway coverage and then formulate goals for the detection of individual actions. It is also preferable, as with the verification goals, to have relatively simple performance objectives from which one can derive specific safeguards implementation plans. A host of uncorrelated detection goals without consideration of complete pathways would be needlessly complex and likely ineffective.

Safeguards measures at declared facilities as well as those designed to detect undeclared facilities both contribute to goal attainment. The identified proliferation activities should be covered via safeguards measures such that the above described performance goals are attained. However, these measures possess important differences. Whereas in most cases the desired timely detection capability at declared facilities can be simply chosen (e.g., via sampling plan considerations), detection capability for off-site activities most often cannot. Therefore, estimates of detection capabilities for undeclared facilities should be used to establish the additional measures necessary at declared facilities in order to achieve goal attainment.

Table 4 extends Table 3 to illustrate how the State-wide performance objectives could be used to derive detection requirements at declared facilities. Two columns are added. The fourth column describes the estimated detection capabilities associated with each type of undeclared facility. The final column shows the additional detection requirements needed at the relevant declared facilities such that pathway deterrence objectives are met.

Pathways involving…	Relative Technical Difficulty	Timely Detection (Deterrence) Requirement	Estimate of Clandestine Facility Detection Capability	Timely Detection Requirement for Related Declared Facilities
Clandestine reactor and/or a	High	Low	Medium	Minimal (defense in depth)
Clandestine Enrichment facility	High	Low	Very Low	Low
Clandestine reprocessing facility	Medium	Medium	Low	Low-Medium
Clandestine conversion facilities	Low	High	Very Low	High

Table 4. Illustrative Derived Timely Detection Requirements for Declared Facilities

For example, perhaps it is determined that detection capabilities for a clandestine enrichment facility, by themselves, are insufficient to meet deterrence objectives. This would suggest that diversion of low enriched fuel at declared facilities should be safeguarded at a level sufficient to meet deterrence objectives. This provides clear guidance for determining inspection objectives.

State-Specific Safeguards Approaches

Performance criteria would enable flexible safeguards approaches based on State-specific issues as well. As noted, pathway analysis is very promising here. Detection probabilities, as well as deterrence objectives

would be assessed. The pathways themselves could be defined and prioritized on a State-by-State basis. Some paths may simply not be credible or relatively more difficult to implement in a given State. A deterrence plan could be tailored to reflect this reality.

Safeguards implementation issues could also be taken into account. For example, if inspectors are given greater access, e.g., on an unannounced basis, other safeguards measures could be used less frequently or even dropped if it could be shown that deterrence objectives are still being met.

A flexible system of this nature would not be discriminatory. Different safeguards plans would likely be implemented in different States but paths of the same priority level would receive the same level of attention. While the issue of determining the proper detection levels for achieving deterrence remains, once set for each category of path it could be applied uniformly to all States.

Conclusions

The proposed performance-based approach to safeguards would provide a more effective means of designing and assessing safeguards systems. By separating and clearly specifying the dual objectives of verification and deterrence, it becomes possible to link specific Agency activities to each goal. Such a construct would simplify safeguards implementation and clarify reporting activities. The strength of any safeguards conclusions would also be more clearly reflected in the performance criteria themselves, giving them tangible meaning particularly in the area of deterrence.

A performance-based system would provide the desired flexibility in the safeguard system. Safeguard approaches would be tailored to State-specific realities, utilizing all available information. This added flexibility will not be gained at the cost of greater discrimination. A clearly stated set of objectives will remain for all States. In any case, without a clear articulation of performance objectives, the effective resource allocation and flexibility desired under integrated safeguards will not be possible.

End Notes

[1] The views and opinions of the authors expressed herein do not necessarily state or reflect those of The Regents of the University of California, the United States Government, or any agency thereof.

[2] The source of these requirements can be traced to INFCIRC/153. In Part I, the basic undertaking of safeguards is described in terms of “verifying that such material is not diverted…” Part II of INFCIRC/153 states, “the objective of safeguards is the timely detection of diversion of significant quantities of *nuclear material* from peaceful nuclear activities…”

[3] Verification also plays an important role in support of international nuclear trade. States desire to have assurances that material and/or technology sold is not used to support a nuclear-weapon program.

[4] In the above example, three levels of deterrence are used. Of course, additional levels could be added if greater fidelity were desired.

National safeguards creates confidence between generations

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Abstract:

The national system is responsible to obtain the necessary information and to create knowledge as well as to take required action or to facilitate effective response. The generation in charge today is accountable to the next generation and beyond. They must create the knowledge, maintain it and share it within and out of the organisation with relevant institutions. In Finland the resources of the national system of safeguards are conscious of that responsibility. The system now in action can be understood as a network of competencies including their functional relations as well as required infrastructures. The aim is at ensuring full compliance with all relevant agreements, such as the Non-Proliferation Treaty (NPT) and the Euratom Treaty. The system resources are active in ministries, industry, R&D companies, customs, police, intelligence organisations etc.

In order to enable effective implementation of IAEA NPT-safeguards measures the SSAC, a dedicated element of the national system, will function as an interface. The Finnish SSAC includes the system of accounting and control as applied by the facility as well as activities performed by the competent authority STUK, Radiation and Nuclear Safety Authority. It is important to note that the undivided responsibility of the safety and security of nuclear materials and activities rest on the operators. The role of the STUK is to set up requirements and to verify the compliance by reviewing and approving the plans and by using different ways and means to verify the performance. The elements of the Finnish national safeguards system have been described in various occasions and papers in the past and it is not repeated here. The focus of the paper is in knowledge creation and sharing processes between the generations aimed at ensuring effective response from the safeguards and security regime.

Keywords: continuity of knowledge between generations, SSAC, NPT, Euratom Treaty

1. Introduction

Global, regional and local safety and security builds upon national, regional and multinational response to dangerous developments. Prevention, detection, corrective actions and other responses to be effective require cooperation. Resources involved in activities represent normally different generations. Confidence between generations is important to facilitate direct and open communication, to ensure critical assessment of the given situation, to create knowledge and share it so as to enable decision and effective action.

Safe-guarding national interest in the use of nuclear energy in Finland has involved different generations. Collaboration has extended over a long period of time from acquisition of initial scientific knowledge about fission late 30's and learning about atomic bomb during the 40's, in the middle of the World War II. A research reactor and the two power reactors from East and West were purchased at the height of the Cold War. Exigencies of 90's and the end of the Cold War as well as the event Finland joined the European Union were further occasions were testing the competencies working for the national safeguards.

It is evident that there is nothing really dramatically new to discover. However, we became aware of something old that from time to time seems to be forgotten. Many times the attention is not managed to that, which would ensure efficient confidence building. Focus is on information systems and actions,

not on knowledge creation and sharing. A precondition for effective cooperation aimed at improving security is not thereby satisfied. What were some of the key lessons learned by the representatives of the two first generations and how the values were transferred to the one now in charge are the main questions elaborated here.

2. The first generation and safe-guards

2.1. Basic motivation

National safeguards in Finland have always been motivated by the national safety and security needs. The early engagement by the end of 30's was mostly motivated by scientific ambitions. Very soon in the 40's due to the atomic bomb direct security relevance of the work became evident. Paris peace accord in 1947stipulated that Finland shall not have nuclear weapons and missiles. It was a conscious decision of the few leading personalities to comply fully with this condition.

This was the key condition, which all Finns had to accept and continue actively developing ways and means to safe-guard the national safety and security interests. Circumstances were not easy. East and West were heading towards a Cold War including rapid nuclear armament. At the same time nuclear energy was marketed as one of the most promising sources of energy. Finland was interested to have a share in that. After the war amount of *fear and distrust* was guiding the people. How conduct in manner which would lead to credible assurances of full compliance and thereby have access to nuclear technology, was the challenge faced by the safe-guards pioneers.

2.2. Need to work on assurances

The pioneers, few individuals working in Finland and collaborating with international partners, had to *take good care* that Finland can be seen by the USA and the USSR as a trustworthy party in the area of nuclear activates, including research. Confidence building was demanded. What guided the few individuals in responsible positions? How it worked and what became possible?

During the 50's and 60's the personalities representing the pioneering generation were trusted to develop conditions that were necessary for the acceptance of nuclear energy production in Finland. This challenge was not only scientific and technological learning exercise but brought along political challenges. Self-confidence and reliable relationships were needed.

There were at that time no large-scale industrial uses of nuclear energy yet. Therefore, no perceived need for a formal national safeguards system existed. Institutional framework was traditional, ministries and research establishments and committees were addressing radiological safety and issues relevant to ensure energy supply under all conditions. The members were few representing the interests of the society. Their knowledge was not limited to technicalities, they managed a socio/technical complex with political, economical and defense implications.

Self-reliance, competence and reliability of these personalities and other involved parties, in and out of Finland, were tested and proven in the middle of the exigencies of the Cold War.

2.3. Events and undertakings offering valuable experiences:

- Scientific knowledge was acquired about nuclear energy, and about weapons and possible consequences, if incidentally or accidentally detonated in its territory. One of the motives was, not to give a reason for the USSR to offer its assistance in this sensitive area, without Finland explicitly asking for it.

- International discourse around "atoms for peace" -initiative was closely followed. Finland joined the newly created IAEA. Not as a small one among the founding members, but the first one to be accepted as a new member - with applause [1].

- In the research area Finland preferred the Nordic cooperation instead of signing cooperation agreements with the big nuclear powers - There was a perceived risk of undesirable dependencies and mistrust developing, as a consequence.

- The research reactor was obtained early 60's from USA and, at the same time, the nuclear fuel for the sub-critical assembly purchased from the USSR via the IAEA. - An opportunity for the USSR to supply nuclear fuel to a western country. PWR from the USSR and a BWR from Sweden were obtained by the end of 60's, after quite complicated political and economic bargaining process.

- In the general conference of the IAEA that had on the agenda the possible role for the IAEA in the implementation of the NPT. Finland used the given opportunity and took the floor to speak first. In its short statement Finland stated that it did not see any need to create a new organization for that very purpose. - It had already used to work with the IAEA.

- Between the International Atomic Energy Agency (IAEA) and Finland fhe first INFCIRC/153-type safeguards agreement (INFCIRC/155) entered into force on 9 February 1972. "Finland was eager to conclude its safeguards agreement as soon as possible. On 11 of March the Director General sent a circular letter to Member States inviting comments on a draft of a model NPT safeguards agreement that the Secretariat had prepared after exploratory discussions with the Finnish delegation". This way David Fisher describes the role of Finland in the early days of NPT safeguards implementation [2].

2.4. The next generation was guided with the following:

- Nuclear energy, among other sources of energy, is important for Finland. The associated issues had to be addressed with care. Trust and commitment were values of fundamental importance. Sensitivity was also required. Certain amount of fear and distrust were important guiding in seeking to understand the risks and what was really needed or wanted in a given situation.

- Care had to be taken to timely engage young people. A cap between the generations shall not develop. Encourage them to take full responsibility. Arrange opportunities for them to participate at all levels within and out of Finland. Remember - external justification is good for healthy self-confidence and necessary for assurances.

- Responsibility should not only mean a functional obligation but have an ethical dimension meaning that the one is answerable, accountable to another, for something. One is a being, capable of fulfilling an obligation and trust, is reliable, and is trustworthy.

- Professor Pekka Jauho in his autobiography [3] offers a word of caution "Global information society is already a reality. However, we should be some what skeptical and cautious, it is not at all certain if the outcome of this development would be a knowledge society". Knowledge creation and sharing is however necessary for effective conduct. The man creates knowledge, not an organization. Moral responsibility can not be avoided. Communicating directly with the relevant parties is an imperative in maintaining confidence. Knowledge is an action oriented construct - Information and data are not! Knowledge creation and sharing is about using power, about causing consequences.

- "Despite of all law respecting States joining the NPT, there is one danger left for us to be taken care of, namely nuclear terrorism" writes professor Erkki Laurila, one of our pioneers, in 1977 [4]. At that time 120 States had joined the NPT. Now the number is 189 and we have it.

Time for the pioneers was passing and the new generation took the responsibility early 70's. The pioneers continued to participate in the societal activities in various capacities. They actively followed up the developments in the areas of safety, security and safe-guards. Some new people were charged with the responsibility to establish and implement a national system of safeguards.

3. The second generation and safeguards

3.1. Need to approach safeguards more systemically

When signing the Safeguards Agreement, Finland undertook to establish and implement a State System of Accounting for and Control of nuclear material (SSAC) within its territory, jurisdiction or

control. The task of the second generation was to continue developing the national safeguards in general and, in particular, to develop an interface (SSAC) with the IAEA safeguards system in Vienna.

Experiences gained in the establishment of the national system and in collaboration with the IAEA during the 80's and 90 contributed to the knowledge of the national safeguards.

3.2. Events and issues appearing of particular importance:

- Opportunities were given for the new generation to interact with the pioneering safeguards generation within the IAEA. Many of the staff members of the IAEA were people with personal war experiences, experiences of the Second World War and of the Cold War. The experiences were not limited to war, but were extended to cover political, international institutional and nuclear technological experiences and knowledge.

- Sensitivities and secrecy naturally covered the sharing of the knowledge. The communication was thus constrained. The Secretariat was maintained as a technical organization. Its Board took care of the political pressures. Within these constraints there was transparence and the interaction with knowledgeable open-mined people helped to understand better the international safeguards and expectations to the national system

- The secretariat was a forum where confidence building between generations representing different national and international institutions was possible. This experience encouraged working on further development of the national safeguards to enable through its findings the most effective implementation of the IAEA system.

- IAEA was growing strong in safeguards during early 80'. Interaction and experiences gained in the implementation of safeguards in Finland and experiences within the IAEA suggested that the role of the national system could be questioned? – Should it be only a facilitator of the reporting and the IAEA inspections? Or, should we have a national safeguards system that would play a security relevant role in the implementation?

- Towards the end of 80's the safeguards implementation developed into a routine operation, meaningful interaction was rear, the effectiveness appeared questionable, motivation at the national was in risk. In order to maintain motivation the people took initiative to start the Finnish Support Programme for IAEA safeguards. The third generation was hired and charged to put meaning back to the national safeguards.

- At the time Finland prepared to join the European Union in 1995, a conscious decision was taken to maintain a national system and to strengthen its functions. The decision was significantly influenced by the exigencies of early 90’s, Iraq et.al., and the associated critiques suggesting that the IAEA safeguards is too constrained and implemented too mechanistically. The third generation now in charge focussed their efforts on strengthening the role and re-defining and functional responsibilities of the national safeguards in this new situation.

3.3. The following lessons were guiding the third generation:

- Safeguards are an obligation that expends over the generations. Safeguards measures are aimed at preventing danger associated with use of nuclear materials. Although the danger may not always be perceived to be present now, planning and implementing measures against it are prudent, because of the unacceptable consequences.

- National safeguards are an essential element in taking care of national security and safety of the nuclear activities. The accountability is extended beyond the limits of the State sovereignty therefore cooperation with all regime elements is of fundamental importance. National safeguards are not only international or regional obligation referred to in agreements.

- Proliferation of nuclear weapons is a question of motivation and resources. The actor may be a State or a sub-national one. It may be very difficult to define it. Therefore the responsible ones for safety, security and safeguards shall not feel too confident. Healthy cynicism, *fear and distrust* are important precursors for building and maintaining sound self-confidence.

- National safeguards shall be present in situ where nuclear materials are used and activities carried out. Due to the international networks contact with bilateral, regional and global partners is important. Direct *communication* on practical operative issues as well as on any particular issues of concern shall be used to maintain confidence.

- *Verification,* establishment of correspondence between declarations and the given objective realities, continues to be one of the fundamental premises of credibility in safeguards, the other being competent judgment about the completeness of the overall picture.

- *Transparency* is a key to confidence building, but without constraints, critical supervision it may lead to increasing information requirements without a link to knowledge creation process. The national safeguards people must be on guard particularly at this moment when perception of the realities is changing and new objectives and measures have been agreed, methods and techniques proposed for application and people trained for implementation.

- The traditional safeguards were *constrained* during 70's by the sensitivities relating to technology, commercial activities and secrecy associated with nuclear programs. The attention of the safeguards was rigorously managed on the declared nuclear material in its declared use. Access was limited to predetermined strategic points. During 80's the operations of IAEA safeguards were constrained by internal criteria which predetermined the activities and timing. Good discipline in the implementation is fundamental to confidence building. However, care must be taken that it is not over done. This would alienate the actors from the actual realities, the purposes and objectives of the mission.

- When new methods and technology are imported from context that is very different to safeguards, there is *a risk* that the science and technology be misapplied through lack of understanding the nature of the subject matter. If the subject matter would only be scientific and dealt within the scientific community, it could be expected that the internal critics would take care of the mistakes. This is however not the case in the use of nuclear technology. Many other elements come to play their role. Therefore, trusting too much on professionals and authorities is not always an advantage. The national system of safeguards should use a network of competencies in order to reduce the risk of bringing evidences of non-objectivity and these of incompetence.

- National safeguards must continue its support to regional and international authority by insisting on assessment and approval by technical experts before accepting new systems for implementation. The technical investigators must be open-mined and rigorous. The results, the knowledge gained must be shared and also published. This kind of transparency will maintain the required confidence.

- Findings of the national safeguards that are communicated to regional or international authorities for their use must be reliable. Interaction and communication, which will follow through independent party, like regional and international competent authority' prove or disapprove that reliability. Confidence developing thereby is trust, which is based on proven reliability of the objective elements and the reliability of the parties involved functionally as well as in the judgment.This appears to be the key element in the implementation of State-level safeguards.

4. Differentiating between data, information and knowledge

Systemic approach shall not be reduced to system approach managed by organization internal criteria. In hierarchical context that would probably lead mechanistically run routines aimed at demonstrating mainly management efficiency. Systemic approach would imply knowledge creation processes and knowledge sharing in the decision-making [5]. Differentiation between data, information and knowledge is important: Knowledge is an action-oriented construct. - Information is not!

The basic assertion is that data and information generation and dissemination of these 'facts' as declared and verified truths is necessary but not at all sufficient for the understanding and acting in a confidence building and security relevant manner.

During the 90's in the field of strategic management the attention has been given to knowledge based approach [3], [4]. Knowledge is seen as the most important strategic resource of the organization. The capability and ability to create, integrate and apply knowledge is critical to the development of sustainable relationships and effective action. A learning organization is maximizing its knowledge base, the basis on which good judgment and effective decision is embedded.

A common approach is to equate information and knowledge. Information is understood as a communicable element of knowledge. It is seen as representation of accumulated facts or data. This equation and narrow understanding leads within an organization to efforts to maximize efficiency (maximize data-, information base). As a consequence, this approach directs the resources to information collection, analysis and storage functions. Such an approach and organization strategy implies that the organization defines by it self criteria against which the effectiveness is measured. Organization knows that it is doing the right thing. Management has a reason to feel confident.

Lessons learned during 90's, however, suggest that we must differentiate (see Table I). Otherwise we may ignore in the judgement process some specific knowledge relevant to a particular context and situation. The consequences of such ignorance are known. We may construct many reasons to feel confident, but we are not necessarily feeling confident when acting - Preconditions for cooperation are not satisfied and incentive to use manipulation and force is increasing.

Traditional information actions, analysis, evaluation and review processes are not as such appropriate, not effective in making best use of the knowledge available within and out of the organization. Therefore, parties entrusted to take good care of the security and safeguards matters are encouraged to enhance their judgement processes, including abilities of their organizations and individuals to differentiate between data, information and knowledge, to create knowledge and share it timely.

Knowledge, and having more of it, does not necessarily lead to greater success. Information technology impressed people and organizations operating on the assumption that more information lead to greater success can provide evidences to that effect. Knowledge created by the people shall be applied in the decision making and implementation processes aimed at meeting the mission objectives and serving the programmatic purposes.

Level	**Definition**	**Learning**	**Outcome**
Data	Raw facts	Accumulating truths	Memorization (data bank)
Information	Meaningful, useful data	Giving form and functionality	Comprehension (information bank)
Knowledge	Clear understanding of information	Analysis and synthesis	Understanding (knowledge bank)
Wisdom	Using knowledge to establish and achieve goals	Discerning judgments and taking appropriate actions	More secure life and acting with confidence

Table I. Distinctions between data, information, knowledge and organizational wisdom

Organizational effectiveness is also concerned with offering learning opportunities, with making judgments and decisions intended to change the conduct of an actor that is contemplating to act in an undesirable manner. Knowledge is an action-oriented construct, sharing it, will make an effect. Knowledge implies questions about relevance and consequence – about ethics of responsibility. "The wise see knowledge and action as one" (Bhagavad-Gita).

The actual process of making decisions and acting consequently requires the simplification of information and knowledge. The ability effectively choose and apply the appropriate knowledge in a given situation is determinant to success also in the area of implementing safeguards.

Knowledge based performance appear to require experience-centric information systems implying increasing complexity of the management-system interactions. Instead of having a focus on the criteria driven routines the line-management operating in different roles must have the ability to understand the context of the events, capture the own interpretations, see the implications associated with any specific event, and create actionable insights. - In short, to create situation relevant knowledge and be ready to share it. This means that the information systems must be centered on the line managers to enhance their personal effectiveness in co-creating value within the organization and out in the field. For more information about this aspect the reader may consult the book identified in the reference [5].

5. Appreciation of interaction between generations

Like the previous ones, the third generation in charge by now in Finland has been enjoying the collaboration with the previous ones, has taken the responsibility and is continuing to implement safeguards taking good care of the security interests not only of Finland. Experiences of the third generation would be another story. Their new story allows room for critical thinking also about the old good stories. It also allows room for innovation and openness. Their story will provide guidance for the fourth generation that is being hired. But, this would be a subject for another paper.

6. Closing remarks

Strengthening of the non-proliferation regime has become a central security objective of the whole industrialized world during the 1990's. It is an established Finnish policy to contribute actively to these efforts. Promoting non-proliferation of nuclear materials is part of the overall policy of combating the spread of weapons of mass destruction and their means of delivery. The task is a complex one and calls for generations to work side by side, to communicate and to cooperate. Possibilities, including time for such interactivity, must be arranged and quality of relationships well taken care of. Effective non-proliferation regime, including disarmament is obligations for all generations to respond within and out of Finland irrespective of the future role of nuclear energy in the world wide energy strategies.

7. References

[1] Laurila Erkki; *Atomienergian tekniikkaa ja politiikkaa*: Keuruu 1967; p 203.

[2] Fischer David; *History of the International Atomic Energy Agency. The First Forty Years*: Vienna 1997; p 254 The Safeguards Committee 1970.

[3] Jauho Pekka; *Ensiksi kielsin konditionaalin*: Hakapaino Oy, Helsinki 1999; p 128.

[4] Laurila Erkki; *Ydinenergiapolitiikan harhailut*: Keuruu 1977; p 72.

[5] Rautjärvi Juha, Upshall Ian; *Knowledge Centered Performance Ensuring Security Relevance in NPT Safeguards*: INMM Florida USA, July 2004.

Designing a State Level Approach – Considerations on Criteria and Procedures

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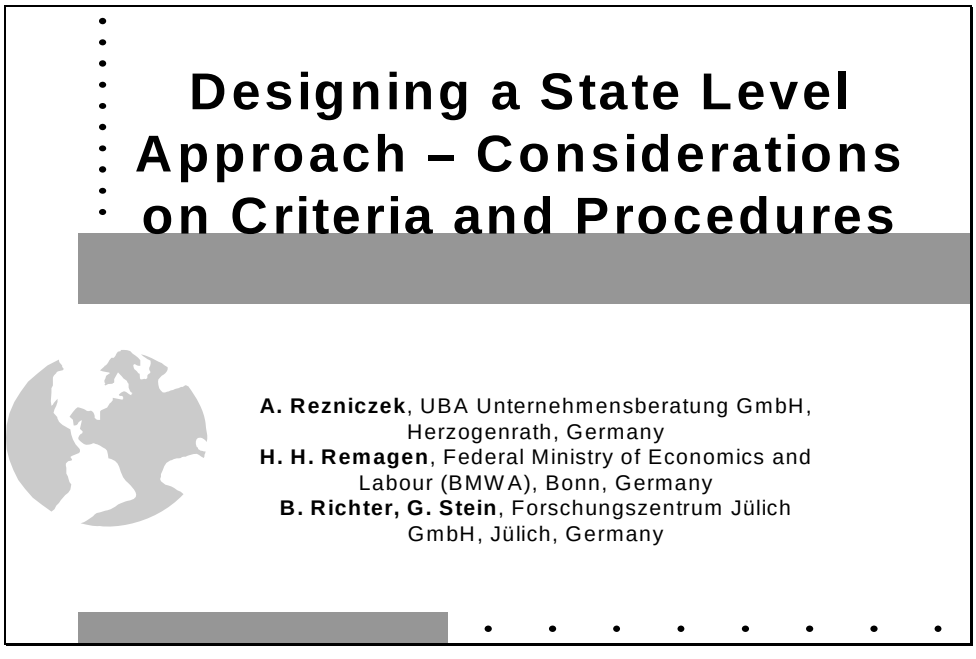
Abstract

The entry into force of the Additional Protocol (AP) gives a strong impetus to the further evolution of the safeguards system applied by the IAEA in the member states of the European Union (EU): the transition from Comprehensive Safeguards to Integrated Safeguards (IS). While some of the new accession states have already implemented IS, the design of IS approaches is still a challenge for the majority of the EU countries and for the IAEA.

A key factor in IS is the shift from a facility-oriented approach to a state level approach. The 'classical' INFCIRC/153 safeguards system puts the main focus on declared nuclear material with a uniform safeguards approach for each facility type, regardless of the state in which the facility is located and regardless of any state specific characteristics. IS broaden the view to the state as a whole and take into consideration criteria like the structure of the state's fuel cycle, R&D activities and industrial capabilities related to the nuclear fuel cycle, the state's commitments to non-proliferation and experiences gained with the state's behaviour in the past.

Within the EU countries, the nuclear industries are characterised by a series of specific factors such as a distinct division of tasks and strong interconnections of the companies across different EU countries, the supervision of national industries and administrations by supranational authorities and a long history of co-operation with IAEA safeguards undertakings. The paper will discuss considerations on criteria to adjust the safeguards approach for state-specific characteristics and on procedures to allocate safeguards effort in a more efficient and flexible way.

Keywords: Integrated Safeguards, State level approach
Number of the theme to which their paper is related: 3 (Safeguards Concepts)
Topic for papers classification: Safeguards Systems - Past, Present and Future



Objectives of IAEA Safeguards

- There are primarily two risks addressed by IAEA safeguards:
  - diversion of fissionable materials from declared facilities (addressed by CSA)
  - and construction of undeclared fuel cycle facilities (addressed by the Additional Protocol)

Integrated Safeguards

- As the IAEA always states, it was never the intention to layer the IAEA‘s safeguards strengthening measures onto one another.
- ‘Integrated safeguards’ are the means by which the IAEA seeks to achieve the most effective and cost-efficient combination of safeguards measures to fulfil its safeguards obligations and meet its verification objectives for states with CSA’s and AP’s in force.

Conclusions drawn from the results of IAEA Safeguards

- A conclusion that there is no indication that a state's declared nuclear material has been diverted from peaceful to proscribed use
- A conclusion that there is no indication of undeclared nuclear material and activities in the state as a whole

Evaluation of a State as a whole

- From all of the information available, the Agency seeks to form a comprehensive "picture" of a State's nuclear programme and nuclear ambitions and identify any potential indications of diversion of nuclear material or of undeclared nuclear material or activities.

What is the Direction for the Evaluation of a State as a whole?

- The procedure is not transparent to the States affected
- In the past, the focus seemed to be on the search for indicators of suspicious circumstances
- Hhow does the IAEA today concider the good references (State specific factors) that could support the assumption of compliance?

State Specific Factors

- INFCIRC/153 Para 81
  - Fuel cycle characteristics
  - International interdependence
  - Effectiveness of the SSAC
  - Verification of flows
- How is this taken into consideration in the evaluation of a State as a whole?

International interdependence

- Multinational Enterprises
  - Multinational staff
  - Production planning and production control under close supervision of all partners
- Facilities owned by non-national companies
- Facilities owned by „Global Players“
- Facilities owned by private shareholders

International interdependence in Research Activities

- Exchange of scientific personnel
- Participation in multinational R&D projects
- Participation in international scientific meetings

Effectiveness of the SSAC

- How independent is the SSAC from the State and operators
- How independent are supervisory bodies in the nuclear field?

Structure of the State

- Number of levels of government involved in nuclear supervision (in federal States)
- How independent are (nuclear) supervisory bodies in a state
- Do independent auditing offices exist

What is a State level Approach

- A State-level safeguards approach specifies the level and focus of safeguards activities for that State.
- Precondition: Evaluation for a state as a whole
- At first State-level approaches were developed fur use under IS, now they will be used for all States with a comprehensive safeguards agreement.

Determination of Inspection Effort in a State for the declared NM

- The approach in the past: Number of facilities based on generic facility approaches
- In a true State level approach not the number of nuclear facilities should be the main parameter to assign inspection effort.
- The amount of nuclear material does not indicate the risk of diversion, the contrary may be true.

Possible Alternative Parameters to determine the Inspection Effort

- What type of measures are necessary to verify that the statements of the SSAC are correct (Verification principle)?
- Modify the inspection effort at facility level according to ownership situation
- Modify the global effort assigned to the State according to the degree of international interdependence of the fuel cycle

Conclusions

- A true State-level approach should consider state specific factors for the total inspection effort spent in the State to verify declared nuclear material.
- Inspection effort provided for in the generic facility approaches should consider facility specific factors, e. g. ownership.

Evolving Safeguards Impose New Demands on Operators

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Abstract

In the field of international safeguards, the operators of nuclear installations in the European Union (EU) have to cope with a number of changes resulting from the entry into force of the Additional Protocol (AP). The AP allows for the application of new safeguards measures by the IAEA, such as environmental sampling, the request for expanded information on activities and sites and complementary access to buildings on sites. Other measures may be implemented in the near future, like unannounced or short notice inspections. At the same time, the safeguards system of the European Community is under revision. The plant operators anticipate drastic changes the consequences of which still have to be investigated in detail.

Hitherto, the screenplay for IAEA safeguards inspections in EU countries was well known and regulated. Article 14 of the protocol attached to the Verification Agreement (INFCIRC/193) states that IAEA inspections shall be carried out simultaneously with inspections of the Commission of the EU. The plant operator was able to rely on the EU inspector to protect his interests in case of disputes during inspections. It was not necessary for him to have a detailed knowledge about rights and duties of the parties in inspections. He communicated in his national language with the EU inspector who in turn carried out the inspection activities together with the IAEA inspector. This situation will change, at least in countries where the inspectorate of European Commission plays the role of a national safeguards system. From now on, the plant operator must train his own staff to deal with IAEA inspections without support from EU inspectors. For most of the measures implemented through the AP, the operator has no practical experience and no detailed background knowledge. To successfully implement these measures in the nuclear installations additional extensive training and exchange of experiences will become necessary. Similar considerations may apply to future EU inspections, if the European Commission shift the emphasis of their inspections to other areas like quality control of data generation within the operator's nuclear material accountancy and control system. The paper examines the new demands the operators will have to cope with.

Keywords: Integrated Safeguards, Complementary Access, Managed Access
Number of the theme to which their paper is related: 3 (Safeguards Concepts)
Topic for papers classification: Safeguards Systems - Past, Present and Future

Introduction

We recognize that international safeguards system is evolving. After a period of stable and established conditions for the conduct of safeguards measures, the operators of nuclear installations within the European Union are facing considerable changes they have to adapt to. Requirements for changes are coming from different sides. The Additional Protocol has entered into force and makes demands on operators regarding new declarations about sites and activities and to prepare for the application of new safeguards measures like complementary access (CA).

The safeguards service of the European Commission also went through a period of change in the last years that comprised a fundamental re-organisation of the Euratom Safeguards Office. The safeguards service is now completely integrated into the Directorate-General for Energy and Transport (DG TREN) with a split of the tasks into two directorates, directorate I and directorate H. Further changes that affect the basic scheme of safeguards measures of the European Commission and the conduct of its inspections are in an advanced state of concept completion and announced for implementation. All these changes impose new demands on operators.

Conditions and Ways to Manage Changes

Operators always have to cope with changes stemming from a great variety of causes. In the commercial environment they have to handle the effects of the globalisation of markets. Mergers of companies have become a nearly usual occurrence that deeply affects the structure and the organizational culture of companies. Operators had and have to respond to changing regulatory requirements at all times. The situation of change is well-known situation for an operator; it belongs to his every day business and operators have quite a lot of experience to deal with this situation.

To cope with changes in a successful manner requires a good change management. The first step normally is to analyse and to try to understand the changes and the possible effects they may have in order to plan the response to the changes. In the field of safeguards the purpose of this change management should be to move from one state to another in some structured and organized way, a strategy of trial and error is certainly not the most appropriate way to adapt to changes in this area. The goal is a smooth transition from one state to another in a planned and orderly fashion. To accomplish this, to operator must be able to analyze the effects and new requirements caused by the changes, to identify or anticipate possible problems and to develop workable solutions to adapt to the new situation. It thus becomes clear that the way the operator can respond to changes strongly depends on the way the changes are introduced and implemented by the responsible authorities. A basic requirement is that the operator is given enough time and enough information to develop and implement a way to make a structured and organized transition.

The Additional Protocol provides for quite a lot of new safeguards measures the operator has to respond to. Essential new elements are the provision of expanded information and expanded access to locations. The operator has to submit on regular basis information about the buildings on the site of his installation, information on R&D activities and information on certain activities listed in annex I of the protocol. The IAEA may conduct complementary access on sites and other locations to assure the absence of undeclared nuclear material and activities.

The process to develop and implement the AP took quite a long time. Already in the early stages of the development of the AP the operators were involved in the debate. They could follow the discussion and analyze possible consequences of proposed measures during the negotiation phase. This helped them to acquire the necessary knowledge to understand the motivation and the background of the new safeguards measures introduced by the AP.

Also the preparation for the implementation of the AP was carried out as a process of discussion and consultation between all parties involved. Again, this helped the operator to analyze and understand the required changes and to organize the implementation of the new measures.

From the side of the European Commission, operators are exposed to changes too. The Commission’s safeguards service were restructured, the new safeguards regulation came into force and the Commission announced new procedures about how to carry out its safeguards obligations. The condition for the operator to install a successful change management are much more difficult in these cases. The restructuring which affects the interface between operator and safeguards services and the development of the new safeguards approaches were carried out without any involvement of the operators. With regard to the new regulation, the timetable for its entry into force had to be adapted several times. The new safeguards approaches were at first developed internally without the involvement of other parties concerned. The lack of more detailed and reliable information makes it at present impossible for the operators to carefully analyze the possible effects of the changes to be introduced by the new approaches and to help to ensure their effectiveness.

Demands and Challenges the Operator is Facing

Fundamental changes in the former well established safeguards situation are occurring and emerging. The operator has to take care of the possible effects these changes may have to his installation and

has to prepare himself to respond to the changes in an adequate manner. There are several key areas in which new demands and challenges can arise.

Departure from Simultaneous Inspections?

Hitherto, the safeguards inspections of the European Commission and the inspections of the IAEA were carried out simultaneously. The safeguards agreement of the EU (INFCIRC 193) provides that the Agency shall apply its safeguards in such a manner as to enable it to verify findings of the Community's system of safeguards. The protocol associated with the agreement states in its Article 14 that the Agency inspections shall be carried out simultaneously with inspection activities of the Community. The agreement also calls for that the Agency and the Community shall co-operate and avoid unnecessary duplication of safeguards activities. Through the years this co-operation developed and led to the "one-man-one-job" principle and was enacted in the New Partnership Approach (NPA).

In practical matters, the close co-operation of the inspectorates leads to essential advantages for the operator. The "one-man-one-job" principle assures that inspection effort can be minimized avoiding that the different inspectors repeat the same work. The simultaneous inspections can ease the communication of the operator with the IAEA inspector. As in Germany for instance, the inspector of the European Commission also acts as the national inspector, he becomes the contact point to whom the IAEA inspector addresses his questions and requests. The operator mainly communicates with the inspector of the Commission and can rely on him in cases were disputes have to be settled.

The new safeguards approaches of the Commission provide for a reduced number of inspections to be carried out by the Commission's inspectors. Compared to the number of inspections under the IAEA's present safeguards scheme, the number of the inspections conducted by the Commission under the new safeguards approaches will decrease considerably. At present, it is not yet clear to what extent the principle of simultaneous inspections will be affected in future. Since both, the IAEA inspector and the operator, rely on the presence and contribution of the inspector of the Commission during the inspection, it will have serious effects if the EC inspector does not participate in an inspection.

If the principle of simultaneous inspection would not be used in future, this could lead to the situation that IAEA and the Commission carry out their inspections at different times. This would increase the burden for the operator with regard to his effort required to prepare for inspections and to accompany the inspectors.

In countries like Germany, where the inspector of the Commission also acts as the national inspector during the IAEA inspections, this role had to be taken on by somebody else. It would mean that additional personnel must be made available for inspections; that this personnel has to be trained in safeguards including the measured provided for in the AP and has to be instructed for this job. Last but not least this personnel must be able to communicate with an IAEA inspectors who does not speak the local language.

A departure from simultaneous inspections would, at least in some countries of the Community, have serious effects on the operator's effort and procedures to prepare for inspections. It is indispensable that the operators have to be involved in the discussion and the further development of the new safeguards approaches. They must be well in advance aware whether this situation will occur and when it might occur to have sufficient time to implement the necessary changes.

Unpredictability

One essential element within the measures implemented through the Additional Protocol is the element of unpredictability introduced into the safeguards measures. The advance notice period for Complementary Access is only 24 hours or, when conducted during an inspection, even only 2 hours. Integrated Safeguards approaches may include unannounced inspections (UI) as a main element.

From the Agency's point of view, the unpredictability aspect of inspections should be very appealing, as long as truly unannounced inspections can be carried out. The possibility to perform inspections or complementary access with only a short time of advance notice or even totally unannounced places the potential diverter in a permanent state of uncertainty and can be used to detect and deter from undeclared activities in a facility and, thus, can be an efficient and cost effective tool to cover a range of diversion scenarios.

From the operator's point of view, it is his main task to run the plant in a safe and cost-effective manner. His planning and management practice aim to reduce all elements of unpredictability and surprise as far as possible to meet his operational goals. For him, the element of unpredictability in the new safeguards measures and concepts is a potential source of friction. There are many ways this can interfere with his operational planning.

Unpredictability is a characteristic of Complementary Access. The operator receives the notification that the IAEA wants to perform a CA at most 24 hours in advance. If CA is to be performed during an inspection, the advance notification time is 2 hours. After this notification time has passed, the inspector is entitled to have access to any building on the site. In many installations, not every staff member has the right to access any place of the installation. The security concept often only provides access to locations where the staff member needs to go to perform his work or duties. Organisational precautions are required in those cases that the inspector and his escort can have access to all buildings required.

Depending on the location, also other departments have to be involved, i. e. the health physics or radiation protection department. For locations where a radiation background has to be expected or for locations that are only sporadically used and therefore not always ventilated as needed, the prior permission of the radiation protection department may be needed. This means, a qualified radiation protection officer has to be available to perform the necessary checks before access is granted.

To provide a suitable escort to an inspector performing a CA also is not a trivial task. The escort must be able and qualified to communicate with the inspector and must be able to accompany the inspector to the desired location. If for example the inspector wants to access a location with a significant radiation level, i. e. spent fuel storage building, this may considerably narrow the choice for the escort. The operator is responsible that the allowed annual dose rates for his staff are not exceeded. He of course wants to minimize the radiation exposure of his personnel and may need the remaining allowed dose rate of qualified staff for future urgent work. People needed to escort the inspector may also be planned on other task that cannot be delayed or whose delay will cause high costs to the operator. Within the narrow allowed time slot the operator has to find a suitable solution.

At a first glance, it may seem to be a trivial task to escort the inspector on the spot to any place on a site, but depending on the type of installation and on the concrete circumstances it may require a lot of preparatory work beforehand and flexibility in the actual situation from the side of the operator to have the necessary people available on a short term basis.

New safeguards measures

The AP provides for a set of new measures. It seems to become a routine requirement that IAEA inspectors take photographs during Complementary Access. Besides his interest to protect commercially sensitive or proliferation sensitive information, the operator has to take into account security or physical protection issues that could be affected by unchecked diffusion of optical information. This means that appropriate rules have to be developed to allow the inspector to fulfil his task while protecting the legitimate interests of the operator. This involves preparation, training and in many cases the presence of a physical protection officer to take and / or check the photos the inspector desires.

Another new measure of the AP is environmental sampling. If the results of the samples show unexpected results, the operator has to investigate the possible causes. To prepare for such a case, the operator should document the location and circumstances when the sample was taken and should consider the necessity to have a reference sample at his disposal. When samples have to be taken out of the facility, this may need the involvement of the radiation protection department to check if the material can be released.

Conclusions

Operators are very well aware that non-proliferation and international safeguards are very sensitive political issues that affect the public perception on the acceptance of the use of nuclear energy and thus are of major importance for the future of the nuclear industry. They are willing to contribute their best effort to establish effective and efficient safeguards. But to be capable of doing this they must be provided with the necessary information and time to adapt to the changes.

Many of the new demands that may look trivial at a first glance require careful preparation beforehand, the involvement of quite a number of staff members with different functions during the execution and flexibility in the actual situation. It is expected from the operator to provide the necessary support that the new tools and measures can be carried out properly and thereby to help to ensure their effectiveness. An indispensable precondition to be able to meet these expectations is the timely information and involvement of the operator already during the early stages of the development of the new standards and approaches. The operators need sufficient information and time to analyze the possible effects of system changes to implement the appropriate reactions and procedures in their facilities.

The Role of Member State Support Programmes in the Development of Technologies for the IAEA's Department of Safeguards

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Abstract:

For over 25 years Member State Support Programmes (MSSPs) have funded research, development and implementation support work for the IAEA's Department of Safeguards. Currently there are 18 MSSPs with over 200 projects spending approximately $20 million/year.

This paper will address the history of these programmes, their changing role through the years and future trends. For many years MSSPs were providers of R&D work but with Agency budget restrictions the work was refocused to address safeguards implementation issues. It is now timely to work on the Agency's future development needs and currently efforts are being made to stimulate R&D activities in the development of novel technologies for the detection of undeclared activities and equipment development needed to support integrated safeguards.

As well as addressing the work performed by MSSPs on behalf of the Department of Safeguards, the paper will also comment on the improvements the Agency has made to its internal systems for planning, requesting and reporting on such work. It will also address changes to be made in the near future.

Finally the report hopes to stimulate debate on the planning and support of long term research and development for international safeguards. The importance of cooperative programmes will be emphasized. These issues are addressed in strategic plans that will be referenced in the paper.

Keywords: IAEA, Member State Support Programmes, R&D, undeclared activities, safeguards equipment

1. Introduction to Safeguards Member State Support Programmes

Member State Support Programmes (MSSPs) voluntarily fund research, development and implementation support work for the IAEA's Department of Safeguards. The objective is to strengthen international safeguards through improvements in the effectiveness and efficiency of safeguards implementation by transferring technology and expertise from Member States to the IAEA.

MSSPs are required because the Department of Safeguards does not have the resources or ability 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 2001-2005, i.e. Objective C.1: To provide greater assurances to the international community that countries are fulfilling their non-proliferation commitments.

In order to obtain the required resources and ability, the mechanism of MSSPs was established. This gives the Department the ability to request, in a uniform and consistent manner, the resources from MSSPs required in order to strengthen the international safeguards. MSSPs accept or reject the requests based on their own resources and governmental policies. The Department then receives the

transfers of technology or expertise either by financial contributions or in kind. It is recognised that there is not an open market for the resources required by the Department of Safeguards to fulfil its mission. For example, much of the training required by the IAEA's safeguards inspectors cannot be purchased. It can only be provided in-kind by certain Member States. The MSSP mechanism is the way in which this support is solicited and delivered.

The first MSSPs were established in the 1970s. There are now 18 MSSPs. These are;
Argentina, Australia, Belgium, 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, Sweden, the UK and the USA.

They contributed, in cash and in kind, over $20 million in 2004 to the Department of Safeguards. This level of contribution has slowly increased with time.

The range of work supported is wide and includes *inter alia*:

- Equipment development
- Training courses
- Open source information collection
- Software development
- New safeguards concepts
- Radiochemical analyses

The work is performed through a number of projects (called tasks) mutually agreed with MSSPs. There are currently about 200 tasks. The tasks are defined, planned and managed by the Department's staff with MSSPs supplying the expertise and resources to perform the work.

2. Historical Context

Member State Support Programmes have existed since 1977 with the formation of the US Support Programme. Since that time the number of support programmes and tasks has increased to reach the current level of 18 MSSPs (Figure 1) and about 200 tasks.

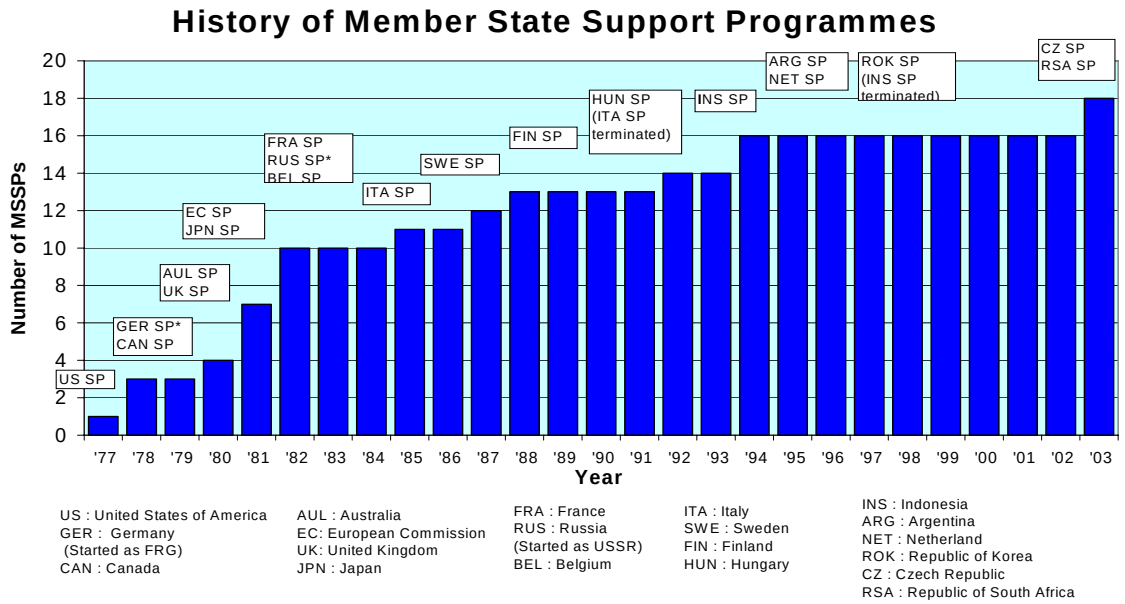

Figure 1: Establishment of Member State Support Programmes

In the period 1977 to 2004 about 900 tasks have been successfully completed. MSSPs have assisted the Department in many and varied ways, perhaps, most significantly, in the area of safeguards equipment and inspector training. The Department currently has an “arsenal” of 100 equipment systems ranging from seals and gamma monitors to sophisticated installed remote monitoring systems. The majority of these systems were specifically developed/adapted to meet the Department’s needs. For inspector training the MSSPs have given the IAEA access to expertise and facilities that are normally unavailable.

2.1. Early Support

Until the 1990s the basic verification measure used by the IAEA was nuclear material accountancy. This relied on IAEA safeguards inspectors making independent measurements to verify quantitatively the amount of nuclear material presented in the states accounts. Much of the early support supplied during the 1980s and early 1990s consisted of the adaptation of existing technologies for use by the IAEA to achieve this purpose. During this period, the principle measurement systems utilising gamma counting (Figure 2), gamma spectroscopy, neutron counting and Cerenkov viewing devices were developed and came into widespread use by the IAEA. These verification techniques were supported by destructive analysis for the detection of bias defects.

Figure 2: Portable gamma attribute tester

Figure 3: Metal seal

Complementary to nuclear material accountancy techniques are containment and surveillance measures, such as optical surveillance and sealing systems (Figure 3). These systems are necessary to assure continuity of knowledge.

By 1990 72% of tasks were related to the development of equipment systems and techniques for non-destructive assay, surveillance and destructive analysis (Figure 4).

The information technology tasks were few and consisted of a number of feasibility studies, database development and the provision of cost-free expert and consultant services.

Safeguards concepts were also being supported but this support consisted of a small number of tasks related to safeguards at new facility types, the dual use of C&S systems and the use of short notice random inspections for inventory change verification.

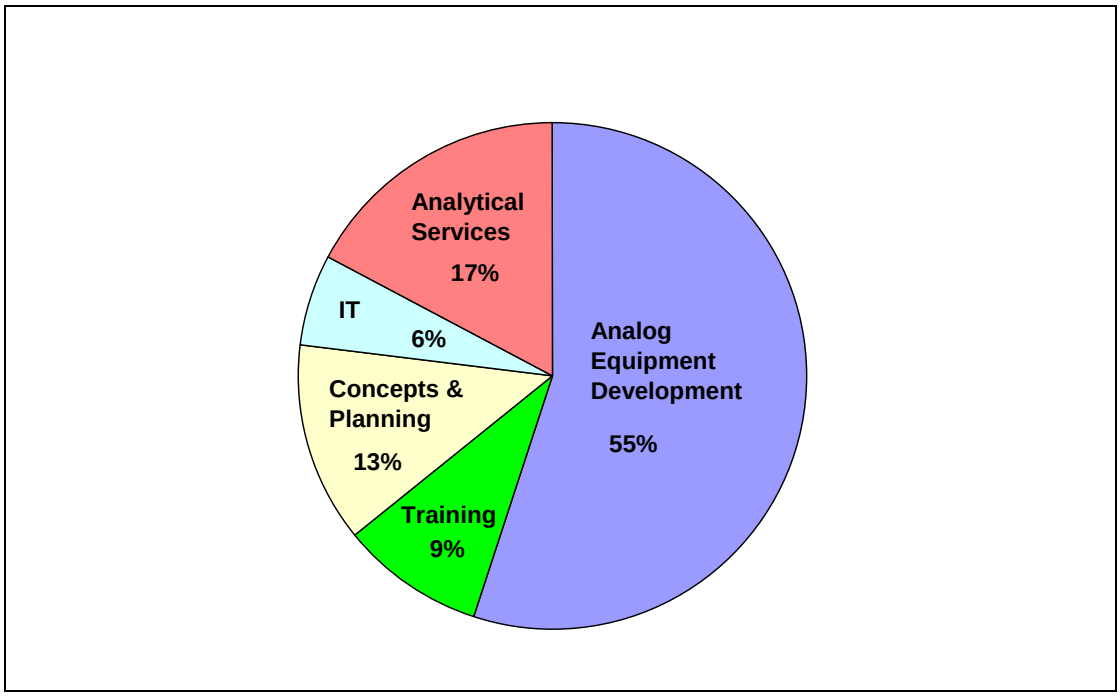

Figure 4: Proportion of tasks by type in 1990

2.2. The Changing Environment

The 1990s saw significant non-proliferation developments and significant technological development changing the emphasis for safeguards development. Significant amongst these were:

- The IAEA's response to the clandestine development of nuclear arms in Iraq and inconsistencies in the initial report from DPRK
- The development of the Additional Protocol and integrated safeguards
- The impact of a zero "real" growth budget
- The availability of digital technologies for equipment systems
- The availability of commercial satellite imagery
- The internet and the ability to collect open source information
- The advances in information technology

By 2000 this changing environment had already had a significant impact on the Department's development programme being implemented with the assistance of Member State Support Programmes (Figure 5). Information technology tasks now totalled 13% of the total number of tasks.

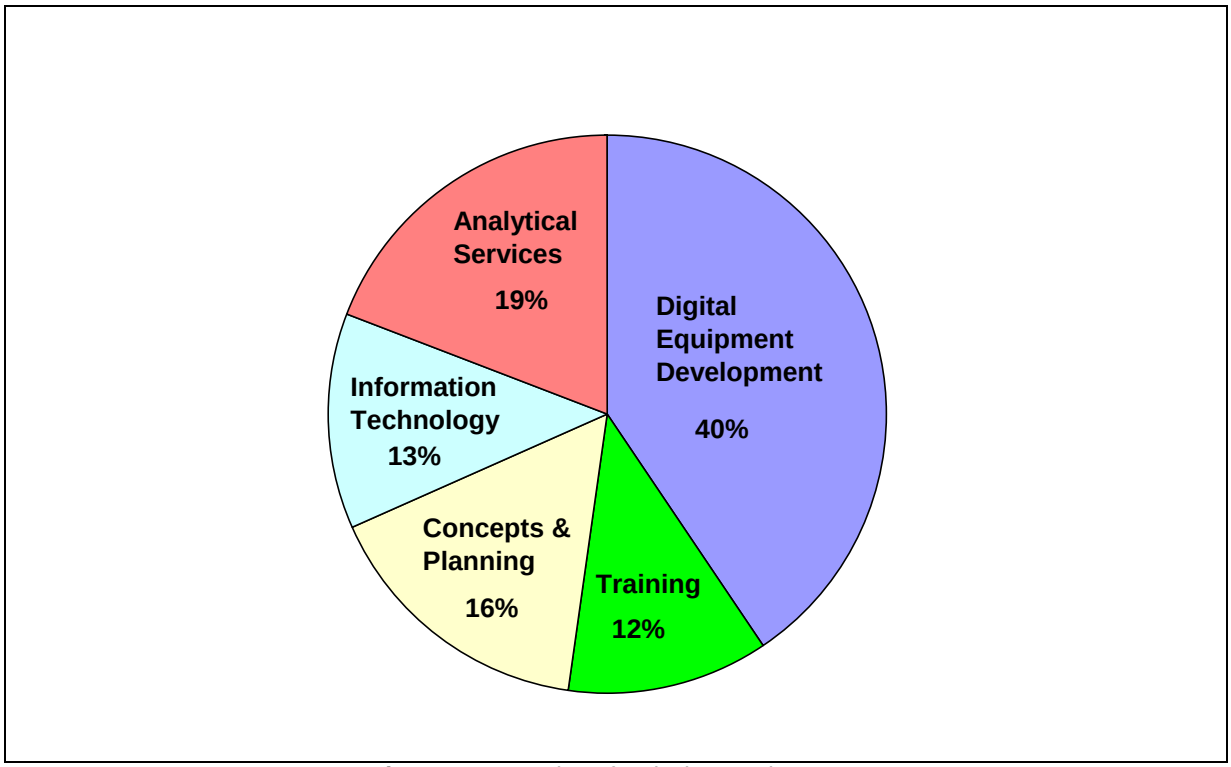

Figure 5: Proportion of tasks by type in 2000

In the area of equipment development new digital surveillance systems were being developed, a new generation of multi channel analysers and gamma detectors (Figure 6) were in the later stages of implementation and unattended, remote monitoring and sealing systems were being designed based on the new digital platforms (Figure 7). The software used to collect and analyse these measurements was being converted for use with Microsoft Windows. Additionally, improvements in efficiency were necessary at this time due to the zero "real" growth budget. The IAEA therefore reduced the number of equipment systems in use and embarked on a programme of standardisation of components.

Figure 6: Portable gamma detector and mulitichannel analyser

Figure 7: VACOSS electronic seal and reader

The safeguards computer network and information systems had matured into a modern network, however, the network contained diverse information systems and databases, each dedicated to specific tasks. Development was confined to the development of these applications as well as the resultant security challenges presented by such a network.

During this time MSSPS also assisted the Department in developing a generic set of concepts and approaches for integrated safeguards. These technical and policy options are being used by the Department to implement integrated safeguards.

3. The Current Position

Since the late 1990s the support for the Department provided by Member State Support Programmes has continued to expand in the area of information technology related work (25% of tasks). This reflects the changing mission of the IAEA and the growing volume and importance of information (Figure 8).

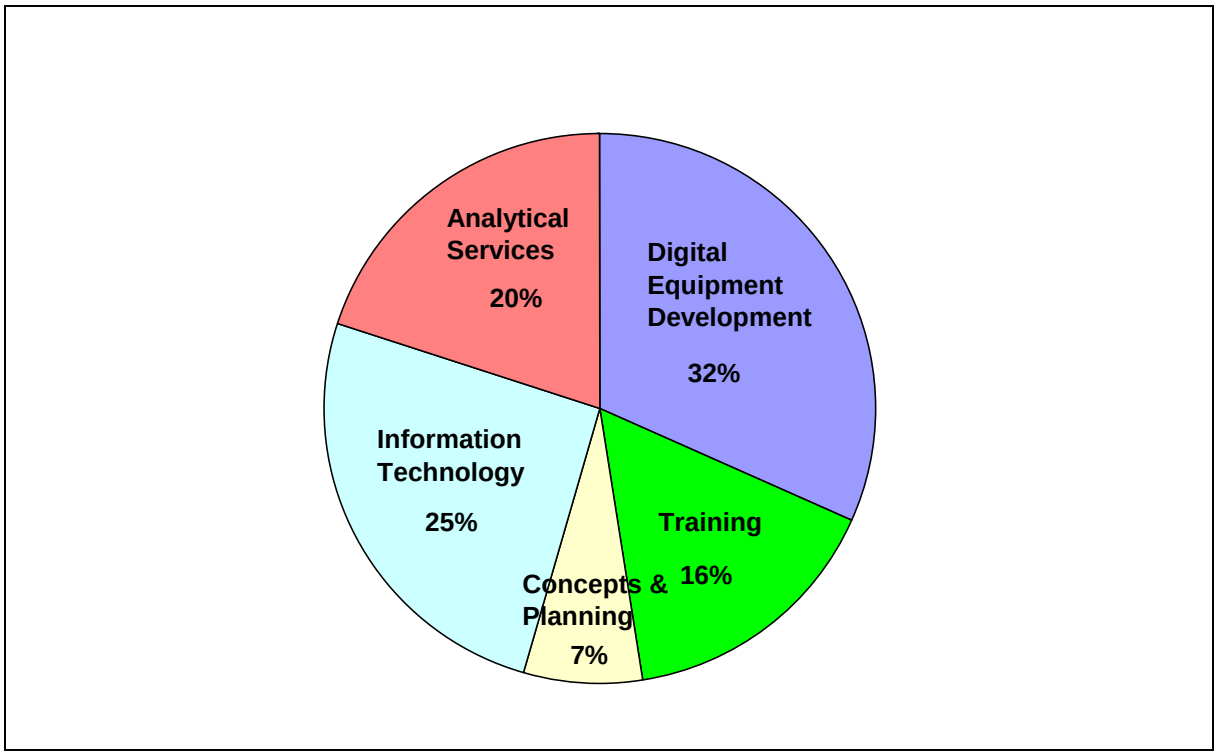

Figure 8: Proportion of tasks by type in 2005

The Department's information technology infrastructure is well developed and work has begun to implement a new $30 million IAEA Safeguards Information System with support from the Member States, predominantly the UK and US. Much of the development work in this area assisted by Member States focuses on the provision and analysis of information for the State evaluation process including open source information, inspection information and satellite imagery.

The Department now has a number of tried and tested equipment components from which it can assemble unattended and remote monitoring systems. Development of these individual components ensures that up-to date technology is employed. The Department is also now looking to the next generation development that will include a new surveillance system. The change in culture required for digital developments has been completed and the development of the new generation is beginning in a systematic manner based on the lessons learnt from the development of the 1st generation digital systems.

With respect to the managerial and conceptual development of the Department consultant support continues to play an important role in this area. MSSPs are also been keen to support the further development of the Department's Quality Management System. Additionally, training tasks remain an expanding need for the Department with the workforce now being trained to reflect the new skills required for Additional Protocol work.

4. Improvements to the Management of Member State Support Programmes

In 2000/2001, as a result of a number of evaluations, the Department of Safeguards initiated a number of improvements including a new system for the approval of support programme task proposals, the implementation of 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 (SPA).

The new system for the approval of support programme task proposals was thought necessary to ensure that no new tasks were started which were not essential to the Department. At the same time the new procedure reduced the number of steps in the processing of task proposals and ensured Departmental review. As well as the new system to approve tasks, the Department implemented a formal project management system for the management of tasks. This project management system was implemented through the new R&D Programme for 2002/3 in which the Department defined projects, appointed project managers and informed MSSPs of the Department's plans.

The R&D Programme is the core planning document in the management of MSSPs. The improvements to this programme and the production of reports based on the programme have brought:

- Centralised planning and coordination. This has resulted in a decreased number of tasks as the Department's priorities are now better defined and redundancy and duplication are removed
- Links to the Department's Strategic Objectives demonstrating clear top-down planning and clear direction
- Decreased workload since the number of administrative steps have been reduced
- Reduced management resources
- Better communication with MSSPs by providing clear up-to-date plans
- Meaningful annual reporting enabling a substantive review of progress and increased emphasis on the completion of tasks

These improvements have been recognised by:

- The external auditor who was "pleased to note that the Agency brought a new system for the approval of Support Programme Task Proposals into effect" and "the involvement of specifically appointed committees and clear definitions of roles and responsibilities improves the accountability process".
- The Standing Advisory Group on Safeguards Implementation who recognised the "very significant improvements in the R&D Programme, which now includes much clearer project management, integration with Member State Support Programmes and better defined project milestones."
- MSSP Coordinators 2003 Meeting at which MSSP coordinators expressed their support for "the robust systems the Agency had installed and encouraged the Department to take full advantage of current systems in expressing the Department's needs".
- The External Evaluation of Major Programme 4, 03-MP-4 dated 5 January 2004 which stated that the "management of Member State Support Programmes has become more focussed ensuring that these programmes support the priority needs of the Department".
- The 2004 Evaluation by the IAEA's Office of Internal Oversight determined that "the Department of Safeguards has achieved substantial improvements in the management of the MSSPs since 2001. The Department of Safeguards has succeeded in implementing the management changes recommended in the assessments and evaluations of 1999, 2000, and 2001. These changes have made a significant impact on the effectiveness of the MSSPs. The Department of Safeguards has improved its R&D Programme, its SPA management processes, and its MSSP task management processes. The new processes are well established, transparent, and facilitate the planning and execution of MSSP tasks."

5. Opportunities for the Future

5.1. New Technologies for Undeclared Activities

At the General Conference the Director General of the IAEA referred to the detection of undeclared activities and said that these "sophisticated and complex technologies will require additional support from Member States – in terms of both technology and budget". The IAEA is reliant on Member States as technology holders to identify safeguards as an application for their existing research and development activities. This statement shows a change of emphasis in the way in which the Department views its R&D activities. In the past the IAEA has made every effort to identify its requirements and to specifically solicit specific contributions from MSSPs. The IAEA now asks the international community to suggest possible new technologies for use particularly for the detection of undeclared materials and activities to assist with the implementation of Additional Protocols and integrated safeguards.

This search for new applications began with the R&D Programme in 2004/5 where an equipment development project was set up to pursue this objective. This project is already showing some promising techniques and feasibility studies are already underway. Such technologies include the laser detection of UF_6 as an indication of undeclared enrichment, noble gas detection indicating undeclared reprocessing and ground penetrating radar to detect undeclared pipes and tunnels.

5.2. Utilisation of Advanced Communication and Information Systems

Technologies already exist to significantly enhance the role of the inspector in the field and to provide mobile communications between inspector and headquarters. The vision of the inspector performing an unannounced inspection carrying a full array of "detection" and information systems has yet to be realized both technically and procedurally but focused development has already started with a feasibility study by the European Space Agency for a secure global communication system. The initiative will continue with a "road-mapping" workshop to be held in the US in Autumn 2005.

A major challenge for the Agency remains the conversion of information and data into knowledge. As each technology is developed and utilized the IAEA finds itself generating ever increasing amounts of data and collecting ever increasing amounts of information. The challenge is to convert this data into knowledge using existing human resources. Development of this knowledge based infrastructure to support the state evaluation process will become a priority for support from MSSPs.

5.3. Management of Technology Transfer

The existing mechanism of using Member State Support Programmes to develop/transfer technologies for use by the Department will remain the main focus. It is a well understood system and has been successful in the past. There are, however, a number of challenges that have to be overcome to progress the development agenda still further. These include:

- Confidentiality and sensitivity in the expression of the IAEA's needs and in the developments being performed
- The involvement of the Agency in international and multinational research networks
- Assistance in developing "non-traditional" technologies
- Recognition by the Agency that Department's resources are required to coordinate and manage development work.

6. Conclusions

Since their conception MSSPs have provided essential support to the Department of Safeguards. This support has contributed to the effectiveness and efficiency of the Department. The MSSPs support has also changed in response to the Departments needs as it moved from a rigid verification based regime into a flexible and more effective regime based on State-level evaluation.

The Department of Safeguards has improved the management of the MSSP resource to better focus tasks on the needs of the Department.

The provision of support, particularly for the development and implementation of new technologies, remains a fundamental part of the Department's resources and demand for support is likely to increase as the rate of technological change increases.

7. Acknowledgements

The authors would like to acknowledge the contributions of Ms. Vera Firbasova and Mr. Kenichi Fujita of the Support Programmes Administration, Department of Safeguards, IAEA.

Session 3

Destructive Analysis

Preparation and certification of IRMM-074, a set of isotopic reference materials of uranium for calibration of mass-spectrometers

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Abstract

A replacement series for IRMM-072 of uranium isotopes ^{233}U, ^{235}U and ^{238}U has been prepared and is being certified. The new series, IRMM-074, follows the design of IRMM-072: the isotopic ratio $n(^{235}U)/n(^{238}U)$ was held constant and the ratio $n(^{233}U)/n(^{235}U)$ was varied from 1 to approximately 10^{-6}.

The isotopically enriched isotopic uranium materials were purified using identical methods involving separation on anion and cation columns followed by a precipitation as peroxide. Each of the three materials were treated in separate clean glove-boxes to eliminate cross-contamination.

The oxides were sintered to convert them to U_3O_8 in parallel in an oven installed in a glove-box that provided a controlled low-humidity environment. ^{235}U and ^{238}U oxides were then weighed together and dissolved in nitric acid. The ^{233}U oxide was dissolved to form a separate solution with the same concentration and from this primary solution 3 dilutions were made by weighing. Weighed amounts of the $^{235}U/^{238}U$ solution and weighed amounts of the original or the diluted ^{233}U solutions were mixed and made up to a concentration of 0.1 mg·g^{-1}. The final solutions were dispensed into individual quartz ampoules that were subsequently flame-sealed.

The uncertainties contributing to the final uncertainties of the isotopic ratios are the weighing errors, the measured impurities in each isotopic material, the stoichiometry of the oxides and the isotopic abundances of each. The main uncertainty is expected to be the measurement of the chemical impurities and for this reason a number of measurement techniques are being employed.

The methods for the preparation and mixing are described and the progress towards certification reported.

Keywords: nuclear isotopic reference materials, uranium, safeguards

1. Introduction

There is a need in Nuclear Safeguards for certified isotopic reference materials of the major nuclear elements of interest: uranium and plutonium. The measurement of uranium isotopic ratios by mass-spectrometry is a special challenge because of the very wide dynamic range of the isotope abundances typically found in natural uranium and uranium in the nuclear fuel cycle. The present effort is a programme at IRMM to replace the set of certified isotopic reference material (IRM), IRMM-072, which was prepared some 20 years ago and which was a set having the highest quality of certified isotopic ratios [1, 2].

IRMM-072 was prepared from purified, enriched uranium isotopic material of ^{233}U, ^{235}U and ^{238}U. The IRMM-072 CRM set was designed in particular for measurement of mass-spectrometer linearity.

Because of the usefulness and general popularity of the IRMM-072 series, it was clear some years ago that a replacement would have to be made and certified. The original methods have proven to be excellent, however where necessary were adapted to present circumstances. In particular the dynamic range of 10^6 was to be retained but fewer members in the set were deemed to be needed. The final results of the new series, Certified Isotopic Reference Material 'IRMM-074', will show what level of accuracy has been achieved.

2. Design of IRMM-074

The design parameters were fixed after many internal discussions at IRMM and taking into account recommendations from the IAEA Reference Materials Consultants Meeting (Vienna, 2002) and from the ESARDA DA Working Group.

IRMM-074 was designed such that each set has 10 individual IRMs; in each of these the isotopic ratio $n(^{235}U)/n(^{238}U)$ was held constant at a value close to unity and $n(^{233}U)/n(^{235}U)$ varied across the series from 1.0 to down to $2.0 \cdot 10^{-6}$.

Following the method previously applied successfully for IRMM-072 the mixtures were made gravimetrically, i.e. by weighing purified, highly enriched oxides into solution and mixing the solutions in the correct proportions, again gravimetrically. The critical points of the preparation were to ensure that:

- Each enriched isotope was handled completely isolated from the others to avoid cross-contamination (to maintain isotopic integrity)
- Exactly the same chemistry purification steps were used (and the same reagents) for each of the enriched isotopic uranium materials in parallel
- The purified isotopes were dried and then sintered together under the same conditions
- A lower concentration for this series than for IRMM-072 (1 $mg \cdot g^{-1}$) was chosen, to conserve valuable certified enriched materials and to better meet the demands of modern mass-spectrometers.
- A minimum weight of each oxide > 100 mg was needed to keep the uncertainties from the weighing procedure as low as possible.

The certified isotopic ratios allow the response linearity of the mass-spectrometer to be determined by measuring the ratios $n(^{233}U)/n(^{235}U)$ and $n(^{233}U)/n(^{235}U)$ across the sequence, each time normalising the response for mass-fractionation by measuring $n(^{235}U)/n(^{238}U)$. The materials can also be used to calibrate the relative response of electron multiplier and Faraday cup collectors for a wide dynamic range of isotopic ratios.

3. Certification of isotopic ratios of starting materials

The isotopic compositions of the IRMM-074 starting materials (enriched ^{233}U, ^{235}U and ^{238}U) were measured using a modified total evaporation technique on the Triton TIMS [3, 4]. Applying the total evaporation technique the measurement is continued until the sample is exhausted. This is done in order to minimize mass fractionation effects. The certified isotopic abundances for the three starting materials are shown in Table 1.

	^{233}U	^{235}U	^{238}U
$n(^{233}U)/n(U)$	99.962 756(47)		
$n(^{234}U)/n(U)$	0.035 908(47)	0.002 128 2(87)	0.000 000 541(12)
$n(^{235}U)/n(U)$	0.000 420 31(38)	99.993 541(21)	0.000 050 7(13)
$n(^{236}U)/n(U)$	0.000 002 535(19)	0.004 228(19)	0.000 000 197 1(73)
$n(^{238}U)/n(U)$	0.000 913 4(24)	0.000 102 38(28)	99.999 948 6(13)

Table 1: Isotopic abundances starting materials

4. Purification and calcination of uranium

Initial experiments were made on the purification of the enriched isotopic uranium materials, using natural uranium for the tests. These investigations looked at the purification of uranium by anion and cation exchange and precipitation.

Each of the steps of the purification cycles was tested by measuring a wide range of impurity elements semi-quantitatively by ICP-MS. The results of these experiments showed that three steps gave an excellent level of purification: anion exchange on BIORAD AG1X4, 100-200 mesh resin in nitric acid medium, cation exchange on Bio-rad AG 50Wx8, 100-200 mesh in HNO_3/THF and precipitation as peroxide.

The enriched isotopic uranium materials were purified using identical methods and chemicals, but in separate clean glove-boxes to eliminate cross-contamination. For each of the three enriched materials, separate glove boxes were installed to eliminate cross contamination or contamination from the environment.

Compared with the previous isotopic mixtures (IRMM-072) higher specification reagents were available and highly pure (sub-boiled or commercially equivalent) water was also routinely available. Moreover, the availability of ICP-MS for measurement of trace amounts of impurities has meant that the purity of reagents and lab-ware can now be controlled much more efficiently.

The purified products were calcined at a temperature of 920°C to arrive at a given stoichiometry in an oven which was installed in a stainless-steel glove-box specially designed and constructed for the purpose; The glove-box environment was controlled so that the humidity in the glove-box and oven were kept at less than 30 ppm throughout the final sintering process. The oven had places prepared for up to four separate oxides. The individual oxides were sintered together in quartz crucibles held in a quartz housing to allow movement of air above the oxides but separating each material from the next.

5. Impurity analysis

The impurities in the enriched isotopic materials were measured by spark-source mass-spectrometry and by ICP-MS. Because of the limited amounts available for ^{233}U and ^{238}U starting materials, samples were only taken from ^{235}U and a test batch of ^{nat}U, which was also used to evaluate different purification methods. Impurity levels lower than 100 ppm were achieved as shown in Table 2.

As the starting materials used for the preparation of IRMM-074, were submitted to the same purification cycle, the assumption was made to assign the same level of impurities for each of the other materials.

	^{nat}U		^{235}U	
	value	Uc	value	Uc
Na	3	<3	0	
Mg	0.5	<0.5	1	0.5
Al	12	10	20	10
Si	6	<6		
P	0.7	0.1	0.5	0.2
S	6	<6		
Ca	8	7	7	5
Cr	1	<0.6	0.7	0.5
Fe	2	<1.5	1.5	1
Cu	2	<0.09	0.1	0.1
Zn	3	0.4	0.3	0.2
Pt			1.5	0.5
Pb	7	3	1	0.5
Total µg/g U	55		36	

Table 2: Major Impurities in ^{235}U and ^{nat}U by SSMS and ICPMS

6. Preparation and Certification

The series was produced by careful weighing of amounts of the two solutions. The mixtures were prepared gravimetrically in quartz flasks, by weighing the purified oxides into solution and mixing the solutions in the correct proportions, again gravimetrically.

In the first stage, ^{235}U and ^{238}U oxides were weighed together and dissolved in nitric acid. The ^{233}U oxide is dissolved separately in a second stage. A mixing schedule as shown in Figure 1has been designed which resulted in 10 solutions covering the range of ^{233}U ratios relative to ^{238}U from 1 to 10^{-6}.

Certification of the isotope amount ratios of the individual units of the set calculated according to GUM[5] and using the GUM Workbench [6].

Four main contributors to the final uncertainties of the isotopic ratio were recognised during the preparation of IRMM-074:

- Uncertainties from weighing: expanded uncertainties (k=2) between 0.012% and 0.024 % on the amount content of the mother solutions of ^{233}U, of the mixture $^{235}U/^{238}U$ and the dilutions of ^{233}U.
- Chemical impurities: based on an equal level of impurities for each of the starting materials as explained before, a value of 100 ppm ± 100 ppm was applied with 0.8 for correlation coefficient.
- Stoichiometry: caused by differences in stoichiometry between the oxides of the enriched isotopic material. For the U_3O_8 a value of 8 was assumed for the oxygen with an uncertainty of 0.01% with a rectangular distribution indicating the limit values for the stoichiometry. A correlation of 0.8 was applied in the calculations.
- Measurements of the individual isotopic amount ratios, expanded uncertainties (k=2) up to $2.6 \cdot 10^{-6}$ % on the molar mass of the individual starting materials were introduced.

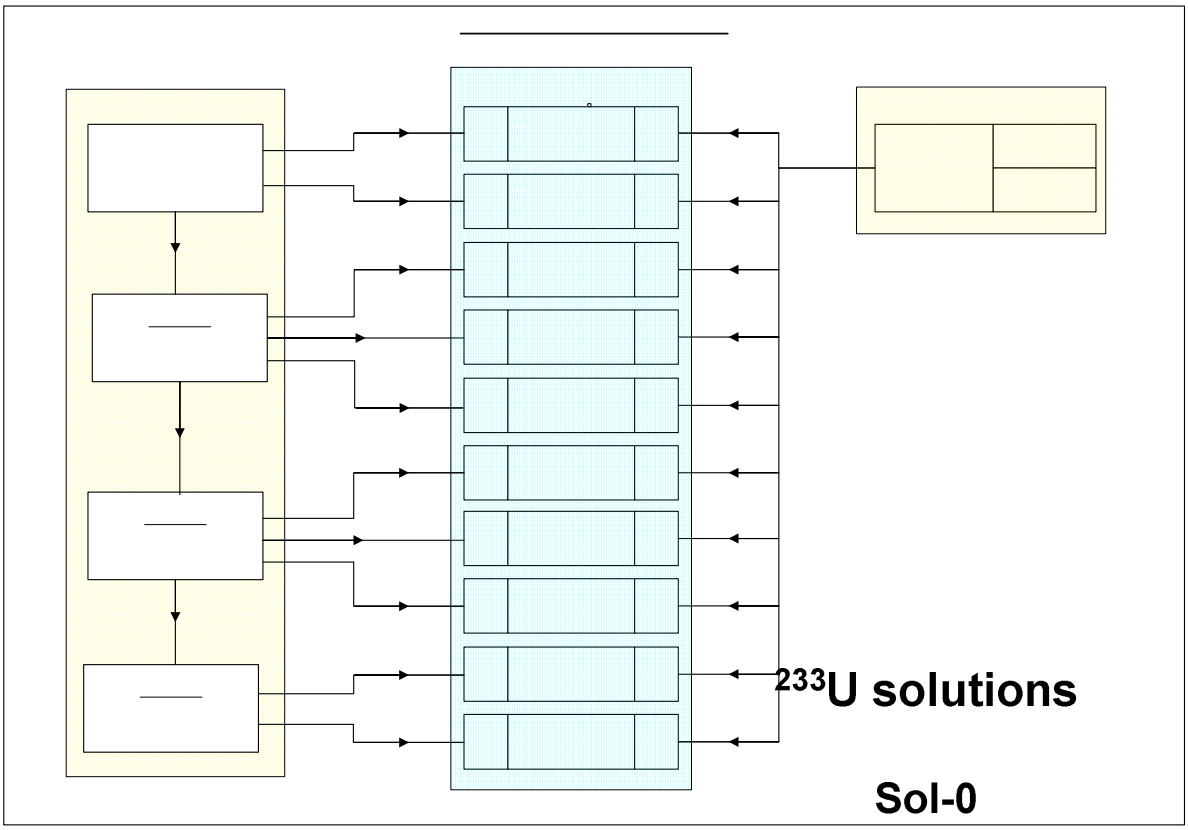

Figure 1: Preparation scheme

The preliminary certified ratios for $n(^{233}U)/n(^{235}U)$ and $n(^{233}U)/n(^{235}U)$ are shown in Table 3 below. Uncertainty budgets with the major components of uncertainty for IRMM-074/1 and IRMM-074/10 is also given in Table 4.

	$n(^{233}U)/n(^{235}U)$	Expanded Uncertainty (%)	$n(^{235}U)/n(^{238}U)$	Expanded Uncertainty (%)
IRMM-074/1	1.014657	0.0098	1.000254	0.0081
IRMM-074/2	0.304337	0.0096	1.000258	0.0081
IRMM-074/3	0.0101494	0.016	1.000259	0.0081
IRMM-074/4	0.00304971	0.015	1.000259	0.0081
IRMM-074/5	0.00102261	0.015	1.000259	0.0081
IRMM-074/6	0.000305	0.019	1.000259	0.0081
IRMM-074/7	0.000101800	0.019	1.000259	0.0081
IRMM-074/8	0.0000305603	0.019	1.000259	0.0081
IRMM-074/9	0.0000081587	0.023	1.000259	0.0081
IRMM-074/10	0.00000101886	0.023	1.000259	0.0081

Table 3: Certified values for IRMM-074 set

Quantity	IRMM-074/1		IRMM-074/10	
	$^{235}U/^{238}U$	$^{233}U/^{235}U$	$^{235}U/^{238}U$	$^{233}U/^{235}U$
Stoichiometry ^{233}U	1.0	0.7	1.0	0.1
Stoichiometry ^{235}U	0.9	0.6	0.9	0.1
Mass starting material ^{233}U		6.2		1.1
Mass starting material ^{235}U	8.2	5.6	8.2	1.0
Mass starting material ^{238}U	8.1		8.1	
Impurities ^{233}U SM		27.8		5.2
Impurities ^{235}U SM	40.9	27.8	40.9	5.2
Impurities ^{238}U SM	40.9		40.9	
Mass mother solution $^{235}U/^{238}U$ (sol0)		5.4		1.0
Combined solutions ^{233}U		8.6		83.7
mass m_{sol3}		8.6		1.0
mass m_{Mix1}		8.6		1.6

Table 4: Uncertainty budget

7. Conclusion

The well-proven techniques that were shown successfully in the preparation of IRMM-072 have been employed for the new series, IRMM-074. The series has been now prepared and first certificate values of the isotopic ratios calculated based on the weights of oxides and solutions and inputting conservative values for stoichiometry and impurities in the starting materials.

Verification measurements are now being carried out and, assuming these show good agreement with the values calculated from the mixing, preliminary certificates can be produced. It will also be important for verification measurements to be made by other laboratories to demonstrate an international validity for the reference material.

8. References

[1] K.J.R. Rosman, W. Lycke, R Damen, R. Werz, F. Hendrickx, L. Traas, P. De Bìevre, “The preparation and use of synthetic isotope mixtures for testing mass-spectrometers”, Int. J. Mass Spectrom. Ion Processes, 79 (1987) 61-71

[2] W. Lycke, P. De Bièvre, A. Verbruggen, F. Hendrickx and K. Rosman, ”A CBNM isotopic reference material set for testing linearity of isotope mass spectrometers (CBNM IRM 072/1-15)”, Fresenius Z. Anal Chem (1988) 331:214-215

[3] Richter S., S. A. Goldberg (2003), Improved Techniques for High Accuracy Isotope Ratio Measurements of Nuclear Materials using Thermal Ionization Mass Spectrometry, International Journal of Mass Spectrometry, 229 (2003) 181-197

[4] Richter S., A. Alonso, H. Kühn, R. Wellum, P.D.P. Taylor (2004), New Procedures for Uranium Isotope Ratio Measurements using the new TRITON Thermal Ionisation Mass Spectrometer, Internal Report IRMM, GE/R/IM/19/04

[5] ISO, Guide to the Expression of Uncertainty in Measurement, 1st edition, 1993, corrected and reprinted 1995, International Organisation for Standardisation (Geneva, Switzerland)

[6] GUM Workbench, Metrodata GmbH, www.metrodata.de

Future Direction of Destructive Analysis for International Safeguards

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Abstract:

Destructive Analysis (DA) for Safeguards (SG) has played a very important role in drawing SG conclusions based on comprehensive SG agreements for several decades. The overall requirements of IAEA DA have gradually changed over recent years, in particular, with the introduction of Strengthened Safeguards. In DA for environmental samples, measurements with better sensitivity and lower background are being pursued, while production and distribution of better QC materials to the IAEA Network of Analytical Laboratories (NWAL) is essential to maintain the highest quality. Further improvement of analytical timeliness is also required for SG analytical services. In DA for nuclear material, a further improvement of accuracy and precision is still a challenge, in particular for verification and accountancy analysis in large scale reprocessing plants, where more reliable and precise reference materials such as spike materials and primary CRMs are needed. More DA capabilities additional to conventional SG analyses should be developed for current and future SG activities, for example during complementary access or inspections under a voluntary agreement. This paper discusses the future directions of DA for international SG based on an assessment of these recent trends.

Keywords: Safeguards, destructive analysis (DA)

1. Introduction

The IAEA uses as one of its basic verification measures nuclear material accountancy, in which IAEA inspectors make independent measurements to quantitatively verify the amount of nuclear materials presented in the State's account. This is in line with Article 55 of INFCIRC/153, which states, "the system of measurements on which the records used for the preparation of reports are based shall either conform to the latest international standards or be equivalent in quality to such standards". Destructive Analyses (DA) combines state-of-the-art determination techniques having the highest possible accuracy, e.g. in accordance with ITV2000 [1], which are then applied for these measurements. This is particularly true for detecting "bias defects" which arise when small amounts of nuclear material are diverted over a protracted length of time. DA involving chemical and physical analyses is the only method to accomplish this important role. DA is also performed for the purposes of verifying the quality and the effectiveness of the Operator's measurement system, as well as for estimating random and systematic errors which are necessary to judge the significance of material unaccounted for (MUF). The Safeguards Analytical Laboratory (SAL) of the IAEA has played this important role of providing DA services in cooperation with a network of laboratories (NWAL) since the 1970s [2]. The realization of timely, accurate and precise DA services within an IAEA On-Site-Laboratory (OSL) at a large scale reprocessing plant is a new challenge for the conventional DA field.

Environmental Sampling for Safeguards (ESS) has been added to the Safeguards DA field since 1996, as a major tool for gaining confidence that illicit nuclear weapon capabilities are not developed clandestinely in contravention of States' non-proliferation commitments. ESS is designed to detect traces of nuclear materials in the environment of a facility that may reveal the presence of an undeclared nuclear activity such as plutonium recovery from irradiated fuel or isotopic enrichment of uranium. Accordingly, the analytical capabilities of SAL have been extended into this area since the

start of its Clean Laboratory's operation in 1996, in parallel with the establishment of additional NWAL capabilities for ESS. Analytical needs coming from 'routine' environmental samples as well as 'special' samples from Complementary Access (CA) or inspections under voluntary agreement (IVA) have remarkably increased in recent years. The analytical requirements for CA/IVA has been extended into 'characterization' of complex sample materials.

In such a context, the goals for DA have been changing constantly over time.

2. Recent remarkable trends and expected changes in DA for Safeguards

Fig.1 represents recent trends in DA for Safeguards, observed over the last 10 years.

The number of samples for nuclear material accountancy verifications has remained constant whereas the analytical requirements for environmental samples, mainly routine samples for ESS, have significantly increased in recent years. The number of screening reports and detailed environmental analyses for particles using Secondary Ion Mass Spectrometry (SIMS) and for bulk samples using Isotopic Dilution Mass Spectrometry (IDMS) is shown in Table 1, where a dramatic increase in analytical samples over the last 5 years can be observed. An increase of hot cell swipe samples from hot cell facilities connected to research reactors is also expected.

So-called 'special' analytical needs, associated mainly with CA and IVA, have turned out to be one of the major challenges for DA in recent years. The number of total special samples in both the nuclear and environmental categories has increased to 271 in 2004 from 177 in 2003. The nature of analytical samples for CA and IVA appears to be a mixture of environmental and nuclear grade materials. DA for investigating the origin of materials will be a new challenge. The analytical requirements for CA tend to be complex; for example, elemental and isotopic determinations (major/minor isotopes) in a variety of matrices (U, Pu, Th, Am, other elements/nuclides), determinations of the chemical form in organic and inorganic compounds. This is already far beyond conventional Safeguards DA and requires stepping into the field of materials characterization.

The analysis of Cm in input and highly radioactive waste samples from reprocessing operations has recently started for the verification of plutonium in hulls and high level radioactive waste in bitumen. Analytical capabilities for measuring Np and Am should be ready in response to the needs for verification of Alternate Nuclear Materials (ANM) [3].

A considerable amount of analytical work for the On-Site-Laboratory (OSL) will be added to the IAEA's DA work in about a year with the start of hot test operations at the Rokkasho Reprocessing Plant (RRP). The OSL is needed to achieve timely analysis without the problem of transporting nuclear samples and to achieve better uncertainties of measurement [4].

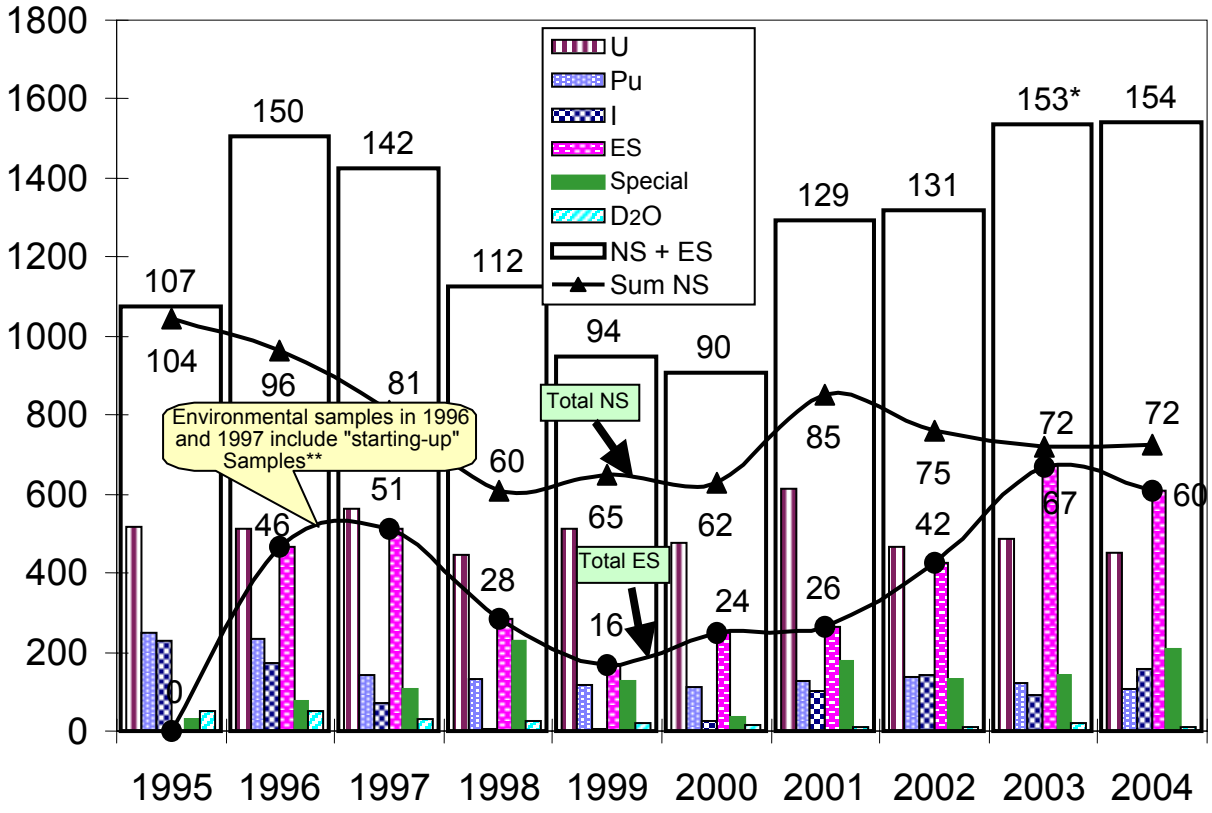
Additionally the IAEA should have analytical capabilities aimed at the new needs for Wide Area Environmental Sampling, which is presently under development.

Table 1 Example of Increase in environmental analysis

Year		SIMS particle	Bulk – IDMS	Screening reports
2000	SAL *NWAL*	23 (32%) *71*	26 (13%) *205*	244
2001	SAL *NWAL*	18 (21%) *84*	20 (7%) *275*	251
2002	SAL *NWAL*	35 (33%) *107*	31 (10%) *323*	327
2003	SAL *NWAL*	98 (37%) *266* Urgent =53%	47 (18%) *264* Urgent =3%	593
2004	SAL *NWAL*	121 (32%) *381* Urgent =75%	70 (20%) *344* Urgent =15%	550

%: SAL/NWAL. ('NWAL' includes SAL)

Fig. 1 Nuclear Material and Environmental Inspection Samples
(Received at the SAL, 1995 to 2004)



NS: Nuclear samples, ES: Environmental samples,
I: Reprocessing input tank samplers,
'Special': Samples from Complementary Access (CA), Mines, other special requests in nuclear material category [special medium samples in environmental category from CA, INVO, air filter are included in 'ES'].
*Excludes 171 INVO old archive samples received.
**Starting-up samples: mainly baseline measurement

3. Current and future needs and goals in DA for Safeguards

Future needs to be addressed in accordance with the above trends are discussed below.

(1) Analysis of 'routine' environmental samples for ESS

It appears that the Agency's analytical regime for routine ESS samples, i.e. 'cold cotton swipes', 'hot cotton swipes' and 'hot-cell swipes', from routine inspection and other types of inspection has been established with the environmental NWAL (Fig. 2) following standardized analytical requests. The following, however, are the remaining respects which need to be improved in the analytical regime for ESS,

i) to establish a comprehensive package of quality assurance (QA) and quality control (QC) to ensure reliability of the analytical results,
ii) to improve timeliness (i.e. achieve faster services),
iii) to equalize the level of expertise among NWALs,
iv) to establish standard guidelines (validated procedures) for each analytical method,
v) to foster more qualified laboratories and more extensive analytical capabilities, and
vi) to have fully independent measurement capabilities within the IAEA.

The improvement of QA/QC, (point i above), in the Agency's environmental NWAL system is underway. Production and use of new QC materials both for particle and bulk analysis have been discussed at various consultant group meetings for ESS held by the IAEA. QA measures to confirm qualitative independence from various interferences in real samples are being studied. The IAEA has so far produced more than 50 kinds of QC control samples for both bulk and particle analysis containing Pu, U (LEU-HEU), Pu+U (LEU-HEU), U-doped glass particles and fission products on cotton or "J-type" cellulose swipes. Further steps include (1) production and distribution of QC samples closer to real samples, i.e., with a variety of environmental dust and (2) systemized measurement of QC samples (with higher frequently) at all NWALs in addition to the laboratories' own QC measurements. More timely analytical services, (point ii above), will be necessary, particularly for certain cases such as IVA where there is a recent trend to increasing the number of urgent samples (see Table 1). The analytical time for measuring and reporting environmental samples needs to be improved throughout the entire NWAL system, although SAL has been striving to meet this requirement. The issues mentioned in points iii and iv, above, are linked with each other. There are significant gaps in the level of quality and expertise among NWAL laboratories, which may partially be due to the fact that state-of-the-art procedures are employed. Establishment of standard guidelines may contribute to improve such gaps. Regarding point v, above, there are considerable differences in the volume (throughput) of analytical services among the NWAL laboratories in spite of the fact that the number of NWAL participants have increased in recent years (as seen in Fig 2). A more even and wider distribution of analytical samples in the NWAL system should be realized by introducing more qualified laboratories and enhancing existing laboratories' expertise and capabilities. A fully independent IAEA capability would be important especially in some very politically sensitive cases. To introduce into SAL certain techniques which other NWAL laboratories possess would fulfill this requirement and may also contribute to faster turn-around for high-priority samples. This would also provide a limited capability for parallel analysis of replicate samples to help assure the quality of the NWAL services.

(2) Analysis of 'special' samples in both nuclear and environmental categories.
This includes samples mainly from CA, IVA and WAES which may be composed of a variety of unknown matrices such as aqueous solutions (acid/ions), organic and inorganic materials, soil, dust, filter materials etc. Such matrices require complex chemistry (dissolution/separation) prior to measurement. Analytical requests may include characterization of samples, determination of concentrations of nuclear materials, other elements/nuclides, isotopic compositions (major/minor) as well as the chemical form (org/inorg) and presence of various anions and cations. This area requires;

i) to establish categorization of samples and procedures for individual categories/samples,
ii) to establish the analytical regime for acceptance of such special samples; i.e. extension of SAL's capabilities and extension of the Agency's NWAL system for this purpose and qualification of its expanded functions,
iii) to establish methods to obtain more information with better sensitivity in lower concentration samples for e.g. origin determination in environmental samples,
iv) to improve timeliness (faster services), and
v) to establish effective QC.

The first should be carried out by accumulation of experiences in analyses of different types of analytical samples. The Agency has carried out many measurements on different matrices, which enable it to make systematic studies. Regarding point ii, some capable laboratories in the NWAL system have been utilized in several special cases, such as for impurity/elemental analysis and analysis of organic materials. Those analytical services, however, have not yet been formally qualified. The establishment of a reliable analytical regime as well as extensive capabilities in SAL are highly necessary. Point iii represents the challenge of obtaining more information from a limited amount of sample, e.g. confirmation of the presence of Pu and Am at lower concentrations than the current detection limit and isotopic and determination of elements other than U and Pu in a micrometer-sized particle. The same discussion as for 'routine' ESS samples is applicable to points iv and v.

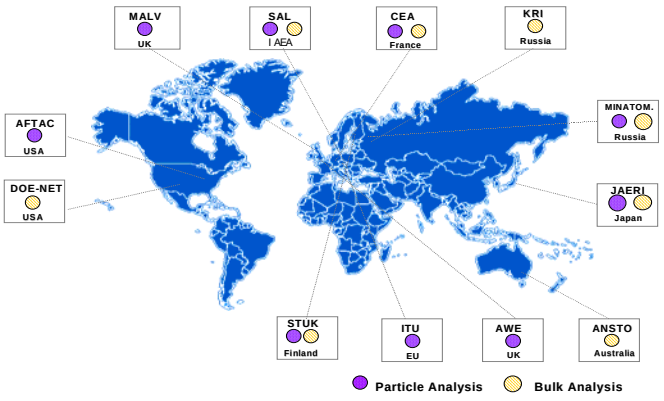
(3) Operation of the OSL
Subsequent to the initial test run with U starting in January 2005, on-site analytical services at the OSL at RRP will begin with real spent fuel toward the end of 2005. The OSL will have to be commonly operated by both the State's and IAEA's inspectorate. Operation of the OSL will enable very timely

analysis without transportation of nuclear samples and may achieve better (i.e. lower) uncertainties of measurement. The following are current issues to achieve success in this project;

i) establishment of operational protocols,
ii) establishment of authentication measures for the first trial of an IAEA/State shared laboratory,
iii) introduction of a QA/QC support regime, and
iv) achievement of better accuracy and precision.

Table 2 shows the analytical methods planned for the OSL, where the methods are classified as to common or parallel use. Operational protocols for either case of common or parallel use have to fulfil the IAEA's requirements of authentication for independence of analysis. Establishment and maintenance of QA/QC in such a remote laboratory is another challenge for the IAEA. Even in such special circumstances, the IAEA may be required to achieve better accuracy and precision in the analytical results because of the higher throughput at a large scale reprocessing plant. Lower uncertainties can be achieved if both the operators' accountancy analysis and the IAEA's verification analysis use the highest quality reference materials [5]; in this context, isotopic spike materials should play an important role because IDMS will be a major analytical technique at OSL. It is therefore essential that reference material producers take up this issue, i.e. to secure provision of such reference materials, particularly large-size dried (LSD) spikes or primary source reference materials with the lowest possible uncertainty, as part of a long-term supply plan.
As technical guidelines referring to the above-mentioned needs, Table 3 and 4 give a typical analytical profile for DA in nuclear and environmental samples [6]. This can also be used as a formal criterion for the qualification of NWAL participants for Safeguards verifications. Sensitivity goals for ESS analysis are proposed in Table 5.

Fig. 2 IAEA-NWAL for Environmental Sampling



MALV:Marvern Technology Centre, **ITU**: Institute for Transuranium Elements, **CEA**: Commissariat a l'Energie Atomique Laboratories, **JAERI**: Japan Atomic Energy Research Institute, **MINATOM**: Laboratory for Analysis of Microparticles, **AFTAC**: Air Force Technical Application Centre, **STUK**: Finnish Centre for Radiation and Nuclear Safety, **AWE**: Atomic Weapons Establishment, **ANSTO**: Australian Nuclear Science & Technology Organization, **SANES**: DOE Network of Analytical Laboratories (ORNL, LANL, LLNL, PNNL, SRTC), **KRI**:Khlopin Radium Institute

Table 2 Analytical methods for OSL: [C]:Common use, [P]:Parallel use

High Active Area	Medium Active Area	Low Active Area	Mass Spec. Area
For input solution and High Active Liquid Waste samples (HALW)	For process solution, output Pu Nitrate solution and MOX powder samples		
HKED[1] [C]	HRGS[3] [C]	U, Pu Chemical [P]	Mass Spectrometry [P]
Separation for IDMS Densitometry [P] Neutron counting (Cm) [C]	Densitometry [C]		

[1] HKED: simultaneous k-edge absorption spectrometry/ X-ray fluorescence technique
[2] IDMS: Isotope Dilution Mass Spectrometry consisting of redox cycle and column separation of spiked diluted samples by a mass spectrometry on a filament
[3] HRGS: High Resolution Gamma Spectrometry to determine the isotopic composition of the samples

Table 3 Profile of Analytical Techniques for Nuclear Materials for Safeguards

Category	Components	Validated Techniques	Combined Standard Uncertainty (1s) [1]	Remarks
Spent fuel	U, Pu elemental	IDMS, HKED	ITV2000	
	U, Pu isotopics	TIMS	ITV2000	
	Np, Am (ANM)	TRU/TEVA separation alpha-spectrometry	5%	(Other Alternative methods applicable)
	Cm	Direct alpha-spectrometry	5 - 10% [2]	For Pu/Cm (in-situ neutron measurement)
HALW	U, Pu elemental	IDMS Spectro-photometry (Pu)	5%	
	U, Pu isotopics	TIMS	ITV2000	
	Np, Am (ANM)	TRU/TEVA separation alpha-spectrometry	5 - 10% [2]	(Other alternatives)
	Cm	Direct alpha-spectrometry	5 - 10% [2]	For Pu/Cm (in-situ neutron
U Product, U process stream (Oxide, UNH, metal, alloy yellow-cake etc)	U elemental	DG-Titration IDMS, Gravimetry	ITV2000	
	U isotopic	TIMS HRGS	ITV2000	
	Impurities [3] (other elements)	ICPMS, WDXRF	5 - 10% [2]	(Other alternative methods applicable)
U-Th Product	U / Th elementals	DG titration +Gravimetry HKED	ITV2000 (U) 0.2% (Th)	Th also possible by complexometric titration
	U isotopics	TIMS	ITV2000	
U Fluoride	U elemental	DG-Titration IDMS, Gravimetry	ITV2000	
	U isotopic	TIMS HRGS	ITV2000	
	Impurities [3] (other elements)	ICPMS, WDXRF	5 - 10% [2]	(Other alternative methods applicable)
U Ore	U and other elementals [3]	XRF HKED	5%	(Other alternative methods applicable)
Pu Product, Pu process stream (Oxide, PuN, metal, alloy etc)	Pu elemental	MS Titration, Coulometry, IDMS, Gravimetry	ITV2000	
	Pu isotopic	TIMS, Alpha-spectrometry	ITV2000	
	Am, Np	HRGS	1 - 5%	
	Impurities [4] (other elements)	ICPMS, WDXRF	5 - 10% [2]	(Other alternative methods applicable)
MOX (Pu/U)	Pu elemental	Titration, IDMS HKED	ITV2000	
	Pu, U isotopic	TIMS, Alpha-spectrometry	ITV2000	
	Impurities [4] (other elements)	ICPMS, WDXRF	5 - 10% [2]	(Other alternative methods applicable)
Heavy Water	Deuterium, Tritium [4]	Densitometry FTIR	0.1% (D_2O) 3 – 5% (T)	
Inorganic Materials	Elemental composition, Impurities, morphology	ICPMS, WDXRF, SEM	5-10%	(Other alternative methods are applicable)
Organic Material	TBP, DBP, Cyanex, others [4]	FTIR, GC, DCI-MS	10 - 20% (or qualitative)	(Other alternative methods applicable)
Acidity	H^+	Titration PH meter (low conc)	1%	
Anion	NO_3^-, Others [3]	Ion Chromatography Ion selective electrode	5 - 10% [2]	(Other alternative methods applicable)

[1] standard uncertainty (1s) associated with the result of one laboratory on a single sample, including all known sources of uncertainty. Numeric value is only relevant for analyte concentration and amount > 5 * L_D (Limit of Detection)
[2] depending on sample size, analyte concentration and matrix
[3] partially unavailable at IAEA-SAL [4] unavailable at IAEA-SAL

Table 4 Typical accuracies for analysis of environmental samples

Measured values	Type/ Technique	Typical accuracy
U amount for 10 ng U	Bulk/IDMS	≤ 10 %
Pu amount for 0.1 ng Pu		≤ 10 %
$^{235}U/^{238}U$ $^{235}U/^{238}U$= 0.001 to 17 for 10 ng U	Bulk/IDMS	≤ 1 %
$^{234}U/^{238}U$ $^{234}U/^{238}U$= 0.00002 to 0.01 for 10 ng U		≤ 10 %
$^{236}U/^{238}U$ $^{236}U/^{238}U$= 0.00001 to 0.01 for 10 ng U		≤ 10 %
$^{239}Pu/^{240}Pu$ $^{239}Pu/^{240}Pu$= 0.001 to 1 for 0.1 ng Pu		≤ 10 %
$^{235}U/^{238}U$ $^{235}U/^{238}U$= 0.001 to 17 for for 1 um particle of UO2 ca 5 pg of U	Particle/ SIMS	≤ 10 %
$^{235}U/^{238}U$ $^{235}U/^{238}U$= 0.001 to 17 for for 1 um particle of UO2 ca 5 pg of U	Particle/ TIMS	≤ 1 %
Fission and activation Products, γ activities (Co60,Zr95,Nb95 Ru106,Cs134, Cs137,Ce144 etc	HRGS	≤ 10 %

Expected blank level
Blank level of NU: 0.1 to 5 ng/swipe
Blank level of Pu: ≤ 10 fg/swipe
Blask level of FP/AP: ≤ 1 mBq/swipe

Table 5 Analytical goals reasonably attainable with state-of the-art techniques for ESS

Analytical method	Feature	Sensitivity goal / Uncertainties
Screening HRGS (HC)	Brief search of Radioisotopes	0.06 Bq ^{241}Am /2h 0.07 Bq ^{137}Cs /2h
Screening HRGS (ES)	Brief search of Radioisotopes	0.03 Bq ^{241}Am /2h 0.04 Bq ^{137}Cs /2h
Screening XRF	Brief search Elements (U, others)	a few ng U/ cm^2 swipe in 100 sec.
Screening TXRF	Screening planchet For SEM/SIMS	10 pg U, Pu/ planchet
Particle SEM	Elemental (U, others) Chemical form of particle	0.1 micrometer particles oxide/fluoride/chloride
Particle SIMS	^{235}U/^{238}U in Rich particles	5% RSD on ^{235}U/^{238}U
Particle FT-TIMS	Accurate isotopics Sparse particles (<20) ^{234}U,^{235}U,^{236}U, ^{238}U	1% RSD on ^{235}U/^{238}U
Bulk AMS	Minor isotope, ^{236}U/^{238}U ^{236}U/^{238}U ratio <10^{-8}	^{236}U/^{238}U ratio <10^{-8} 10 fg Pu with 10% RSD
Bulk TIMS	^{234}U,^{235}U,^{236}U, ^{238}U ^{238}Pu,^{239}Pu,^{240}Pu, ^{241}Pu	100 pg U with ^{233}U spike a few fg Pu with ^{244}Pu spike

4. Conclusion

DA has played a very important role for safeguards verification under comprehensive SG agreements and the Additional Protocol. The overall requirements for IAEA DA have gradually changed in recent years. In particular, the needs for DA in both nuclear and environmental samples have been extended from the determinations of U and Pu to the characterization of sample materials; these are new challenges in DA for SG. Improvement of DA for ESS as regards reliability and timeliness is required. Improvement of traditional DA for nuclear materials by the establishment of accurate and precise analytical operations at the OSL at Rokkasho should be pursued. Flexible management in the field of DA for SG is in high demand to meet these changing needs.

5. References

[1] H.Aigner et al.: “International Target Values 2000 for Measurement Uncertainties in Safeguards Nuclear Materials”, STR-327, IAEA, April 2001.
[2] P.Doherty et al.: NWAL Support to IAEA Safeguards Analysis”, Symposium on International Safeguards, Vienna, October 2001.
[3] Y.Kuno, et al.: ”Development of Analytical Techniques for U/Pu/Np/Am at the Safeguards Analytical Laboratory”, Proc. IAEA Safeguards Symposium, Vienna, Austria, October/November 2001
[4] G.Duhamel et al.: “Establishing the Joint IAEA/JSGO/NMCC Safeguards On Site Laboratory for the Rokkasho Reprocessing Plant: Major Challenge for managing the interfaces” Proc. Int. Conf on Facility Operations – Safeguards Interface, Charleston, USA, February/March 2004.
[5] R.Wellum, Presentation at ESARDA-DAWG, Rez, Prague, October, 2004.
[6] Procedure for Qualification of a Candidate Analytical Laboratory for Agency’s NWAL for analysis of Safeguards Samples, SGTCSP22 Rev.2 (2005)

Evaluating the uncertainties associated with isotope ratio measurements carried out by different mass spectrometry techniques in uranium samples

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Abstract

An experimental programme was set up to evaluate the uncertainties associated with the measurements of the isotope ratio $n(^{235}U)/n(^{238}U)$ by mass spectrometry techniques: inductively coupled plasma mass spectrometry (ICP-QMS and MC-ICPMS), thermal ionisation mass spectrometry (TIMS) and gas source mass spectrometry (GSMS). A group of samples with isotope ratios ranging from depleted (0.5 wt %) to low enriched uranium (3.5 wt %) were selected to enable the comparison of performances of these different techniques. Although all the three techniques provided very accurate results, the GSMS always produced results with the smallest expanded uncertainty. The stability of the ion-beam generated by the electron impact ion source employed in GSMS is regarded as one of the main reasons for this outstanding performance.

Keywords: uranium isotope ratio measurements, gas source mass spectrometry, thermal ionisation mass spectrometry, inductively coupled plasma mass spectrometry, nuclear safeguards.

1. Introduction

Uranium is processed throughout the nuclear fuel cycle under different chemical forms and having different isotopic abundances [1]. During the 20 year span covering the initial mining activity to the nuclear waste disposal, uranium isotope ratio measurements are performed several times for safeguards purposes.

The nuclear fuel cycle is varied and complex and analytical laboratories measure the isotope ratios of uranium samples in the form of solid, liquid or gas. In practice this diversity requires the use of several kinds of mass spectrometers that in turn are affected by different sources of bias and generate measurement results associated with different uncertainty levels.

The aim of this work, therefore, is to evaluate the measurements of the $n(^{235}U)/n(^{238}U)$ isotope ratio performed in a set of uranium samples by the following techniques: quadrupole inductively coupled plasma mass spectrometry (ICP-QMS), multi collector magnetic sector inductively coupled mass spectrometry (MC-ICPMS), thermal ionisation mass spectrometry (TIMS) and gas source mass spectrometry (GSMS).

The main sources of uncertainty in each technique were identified and the values of the expanded uncertainties estimated according to the ISO-GUM and EURACHEM-CITAC guides [2, 3]. This approach allowed a critical comparison on the potentialities of each technique and provided a basis to quantify their main advantages and disadvantages.

2. Experimental

2.1 Materials

A set of four samples ranging from 0.5 to 3.5 wt % of ^{235}U, originally in the form of uranium hexafluoride (UF_6) was selected**.** The samples were distilled into fresh Monel ampoules to remove volatile impurities, especially hydrofluoric acid (HF).

The samples and isotopic reference materials (IRMs) chosen were hydrolysed to uranyl fluoride (UO_2F_2) by the addition of high purity deionised water and then converted to uranyl nitrate ($UO_2(NO_3)_2$) by the addition of Suprapur nitric acid and evaporating to dryness. The final uranium concentrations were adjusted in 1M HNO_3 according to the requirements of each specific technique, as presented in table 1.

Technique	IRM used	Chemical form	Concentration
ICP-QMS	2411,071,295	$UO_2(NO_3)_2$	3.0 ngU/g
MC-ICPMS	021,2079	$UO_2(NO_3)_2$	1.0 μgU/g
TIMS	184	$UO_2(NO_3)_2$	5.0 μgU/g
GSMS	071,021,2079,295,2408,2411	UF_6	-

Table 1: Isotope reference materials and concentrations used in each technique

IRM	Certified isotope ratio $n(^{235}U)/n(^{238}U)$
IRMM 021	0.0044036 (21)
IRMM 2079	0.0071505 (24)
IRMM 071	0.0072623 (16)
IRMM 2408	0.0197333 (52)
IRMM 184	0.0072623 (22)
IRMM 295	0.0307712 (51)
IRMM 2411	0.0406210 (83)

Table 2: List of reference materials used and their certified isotope ratio $n(^{235}U)/(n^{238}U)$. The expanded uncertainty values are presented in brackets and include a coverage factor (k = 2)

2.2 Instrumentation

The features of the mass spectrometers used in this work are briefly presented below.

The MAT 511 is an electron impact gas source mass spectrometer (GSMS) 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 uranium isotopes in UF_6 samples.

The ELAN 6000 is an inductively coupled plasma mass spectrometer (ICP-QMS) manufactured by Perkin Elmer (Norwalk, CT, USA). It is equipped with a 40 MHz free-running radio frequency generator, computer controlled ion optics and quadrupole analyzer. The detector used is a discrete dynode electron multiplier (ETP, Ringwood, Australia). It is also fitted with Ni cones, quartz spray chamber and micro-concentric nebuliser.

The Nu Plasma is an inductively coupled plasma mass spectrometer (MC-ICPMS) manufactured by Nu Instruments (Wrexham, North Wales, U.K). It is equipped with a 27 MHz crystal controlled radio frequency generator, double-focusing magnetic sector analyzer, variable-dispersion ion optics and twelve Faraday collectors. Additionally there are three single electron multipliers (SEM) connected to a retarding potential quadrupole analyser (RPQ), an energy filter used to clean undesired peak tails in low intensity masses and so improve the abundance sensitivity. It is also fitted with Ni cones, quartz spray chamber and a micro-concentric nebuliser.

The Triton is a thermal ionisation mass spectrometer (TIMS) manufactured by Thermo Electron (Bremen, Germany). It is equipped with a sample magazine for twenty-one filaments, 90^{o} magnetic sector analyzer, dynamic zoom optics and nine Faraday collectors, each one with its own signal amplifier. For small signals it also has a SEM device connected to a RPQ.

2.3 Measurement procedures

Each technique has its own procedures which have been optimised for uranium isotope ratio measurements and are fully described elsewhere [4, 5, 6, 7].

2.3.1 ICP-QMS procedure

The ICP-QMS measurement sequence used was the following: blank, IRM, sample. All data was monitored in the pulse counting mode. The instrumental background subtraction and dead time correction were first applied to the raw data and the mass discrimination effect was corrected by external calibration [4]. The correction factor was determined as the ratio between the certified and the observed values of the IRM.

2.3.2 MC-ICPMS procedure

A static multi-collector measurement scheme using two Faraday cups was used. The analysis comprised three blocks with one mass cycle with 30 integrations of 10 s each. The measurement sequence used was the following: blank, IRM 1, blank, sample, blank, IRM 2. The electronic background of Faraday collectors and the procedural blank were measured in the blank. Then they were subtracted from the signal intensities of the samples and standards [5]. Finally the mass discrimination effect was corrected by external calibration using an average mass discrimination factor.

2.3.3 TIMS procedure

Samples and IRMs were prepared in the form of uranyl nitrate solutions at the concentration of 5.0 μgU/g. A drop of 1.0 μL was deposited onto zone-refined rhenium filaments. The double filament technique was applied. The sample drop was dried and the filaments were then carefully assembled in the instrument magazine. The measurements were carried out using the modified total evaporation method [6]. Both samples and IRMs were processed using the same operational parameters. Each analysis comprised 40 to 60 blocks of measurements with 5 mass cycles with an integration time of 32 s. The data acquisition was interrupted regularly to perform focusing, peak centering and Faraday background measurements. The mass discrimination effect was corrected by external calibration. The average mass discrimination factor obtained by measuring several IRMs was applied to correct the observed isotope ratio of the samples.

2.3.4 GSMS procedure

The GSMS measurements were carried out using the double standard method [7], which is based on the bracketing of the sample by two IRMs. The first IRM used had an isotopic ratio slightly higher and the second one an isotopic ratio slightly lower than the sample. The sequence of measurement was executed in two different blocks. In the first, the sequence was: sample, IRM 1, sample. In the second block the sequence was: sample, IRM 2, sample. A mass discrimination correction factor was determined in each measurement sequence. Both factors and the certified isotope ratio of the materials were used to calculate the corrected isotope ratio for the sample.

2.3.5 Uncertainty estimation

The uncertainty estimation was carried out according to the recommendations of ISO-GUM [2] and EURACHEM-CITAC [3] guides. The dedicated software GUM Workbench Pro [8] was used to speed up all the calculations.

3. Results and discussion

3.1 Measurement results

The $n(^{235}U)/n(^{238}U)$ isotope ratio measurements results obtained are presented in table 3.

Sample	ICP-QMS $n(^{235}U)/n(^{238}U)$	MC-ICPMS $n(^{235}U)/n(^{238}U)$	TIMS $n(^{235}U)/n(^{238}U)$	GSMS $n(^{235}U)/n(^{238}U)$
S 1	0.005375 (88)	0.0053559 (25)	0.0053551 (18)	0.0053547 (17)
S 2	0.00725 (11)	0.0072537 (37)	0.0072537 (27)	0.0072543 (16)
S 3	0.02429 (29)	0.024222 (12)	0.0242253 (87)	0.0242320 (42)
S 4	0.03548 (34)	0.0354688 (90)	0.035469 (12)	0.0354698 (47)

Table 3: List of samples and the isotope ratio $n(^{235}U)/n(^{238}U)$ measured by each technique**.** The expanded uncertainty values are presented in brackets and include a coverage factor (k = 2)

The measurement results presented in table 3 reveal a good agreement between all the techniques within their expanded uncertainty (k=2) values. The results of the measurements for one sample (S2) are given in figure 1 showing the excellent agreement between the results.

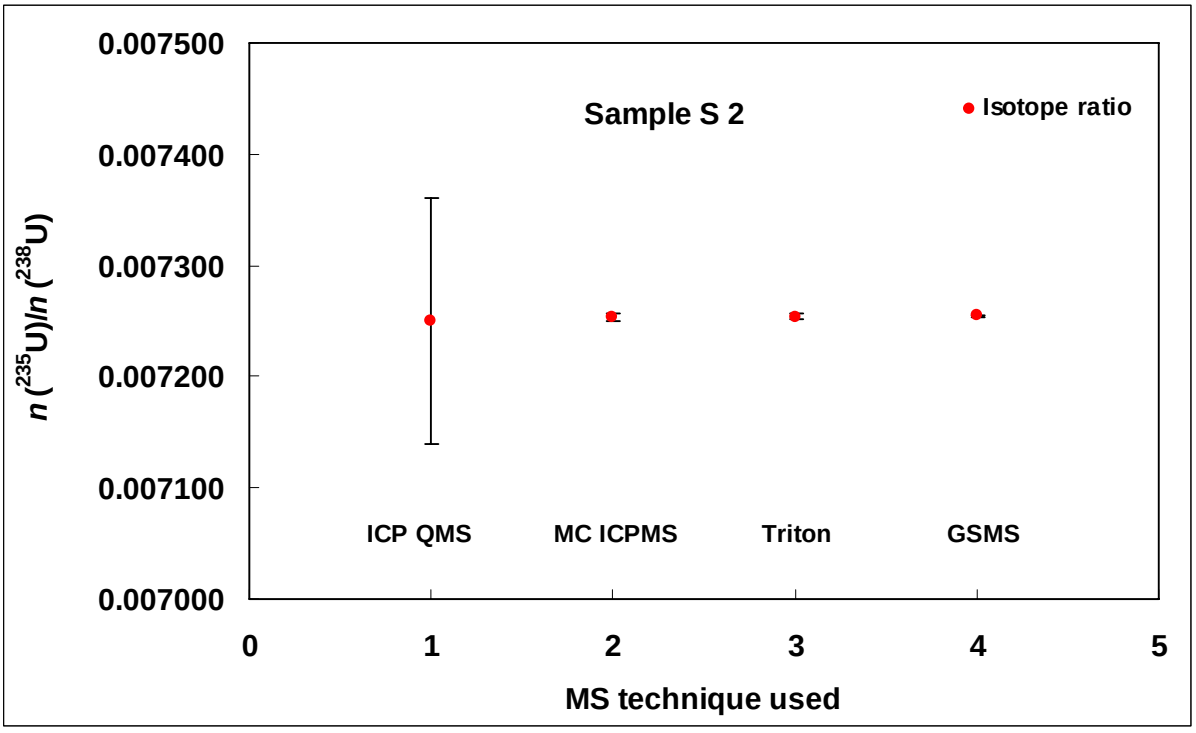

Figure 1: Isotope ratio values obtained for sample S 2

The results from the measurements by ICP-QMS have expanded uncertainties much larger the other techniques. A comparison of the three other methods is shown in figure 2.

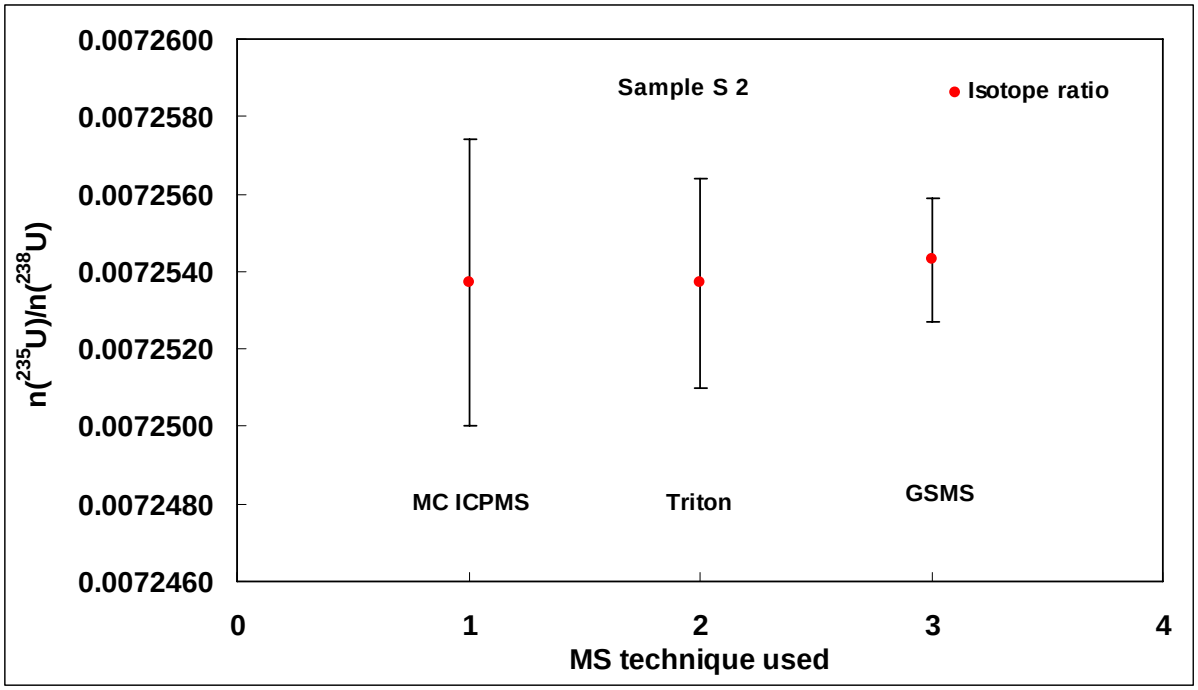

Figure 2: Isotope ratio values obtained for sample S 2 using only the three most precise techniques

The results presented in figure 2 from the three most precise techniques agree well with each other within their stated uncertainties. They also show that measurements by GSMS have the smallest expanded uncertainty. Results by TIMS have uncertainties larger than GSMS but smaller than those of MC-ICPMS. The same trends were found on measurements carried out for the other samples of the set.

3.2 Sources of uncertainty

An uncertainty estimation process was carried out for every sample of the set to identify the uncertainty sources and quantify their contribution to the expanded uncertainty in each mass spectrometry technique. The results obtained for sample S 2 are presented in table 4.

Uncertainty contribution (%)	ICP-QMS	MC-ICPMS	TIMS	GSMS
Reference material	0.0	19.1	66.5	97.4
Repeatability	99.3	66.5	33.5	2.7
Background subtraction	0.7	14.4	0.0	0.0

Table 4: Sources of uncertainty for sample S 2

Table 4 shows that for ICP-QMS, the most important source of uncertainty is the repeatability of the measurements (99.3 %). The other main contribution comes from the background subtraction (0.70 %), which is a demonstration that the memory effect related to the last sample to be processed has always to be accounted for in this technique.

In table 4 the repeatability of the GSMS measurements only contributed 2.7 % to the final combined uncertainty while the reference material contributed 97.4 %. These figures are explained by the high repeatability of this technique, where the relative standard deviation of the uncorrected isotope ratio measurements was about 0.012 %.

Both TIMS and MC-ICPMS exhibited a more balanced performance, with intermediate values coming from these two main uncertainty sources.

The reference material was the main uncertainty source for the TIMS, reflecting the high repeatability provided by its raw isotope ratio measurements whose standard deviation was about 0.025 %.

The repeatability was the main source of uncertainty for the MC-ICPMS which is partly a result of the measurement procedure applied in this case. The standard deviation of the raw isotope ratio measurements was about 0.06 %. It is important to observe that the contribution of the background subtraction was 14.4 %. This means that if this operation had not been carried out, the isotope ratio would be biased and the uncertainty would certainly have been underestimated.

The relative contributions of each source of uncertainty to the standard combined uncertainty of isotope ratios of sample S 2 can also be viewed in figures 3 to 6.

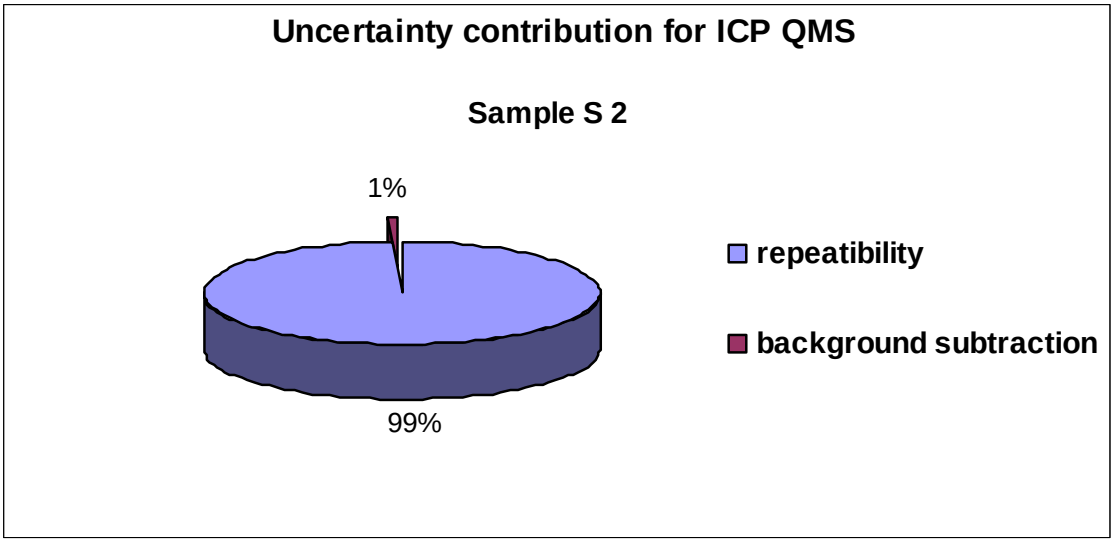

Figure 3: Uncertainty contributions for sample S 2 for ICP-QMS

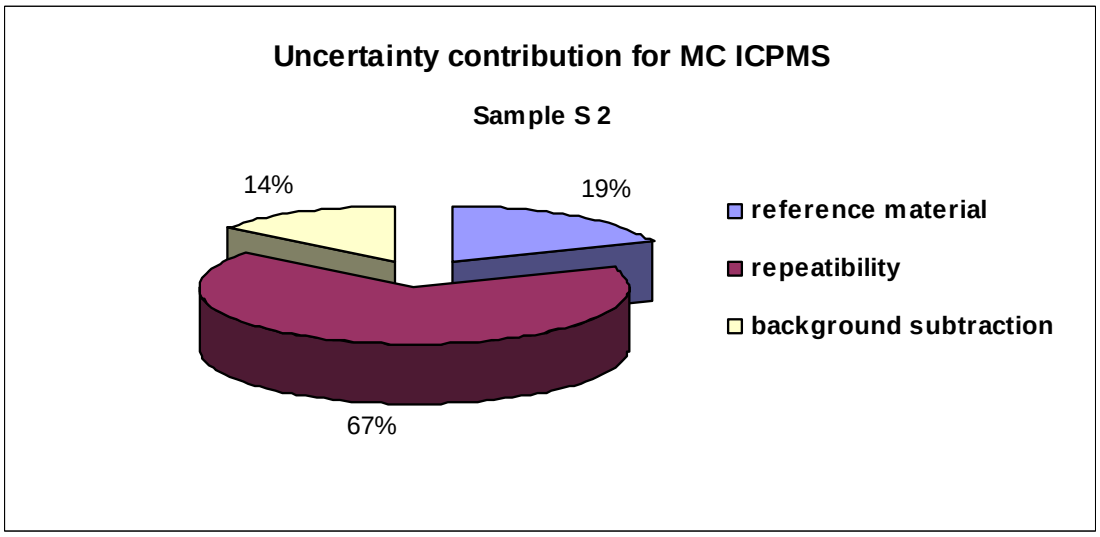

Figure 4: Uncertainty contributions for sample S 2 for MC-ICPMS

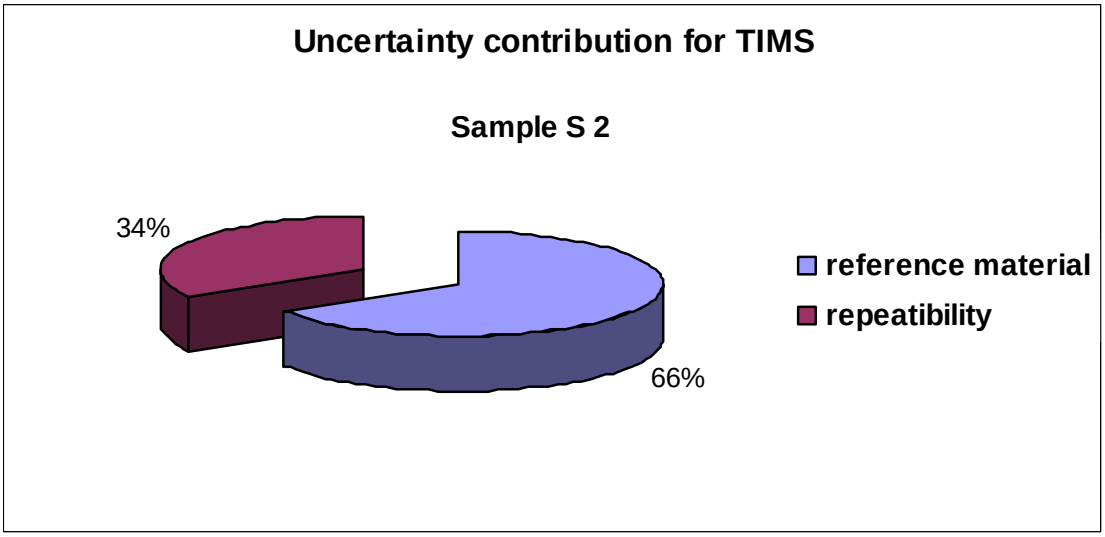

Figure 5: Uncertainty contributions for sample S 2 for TIMS

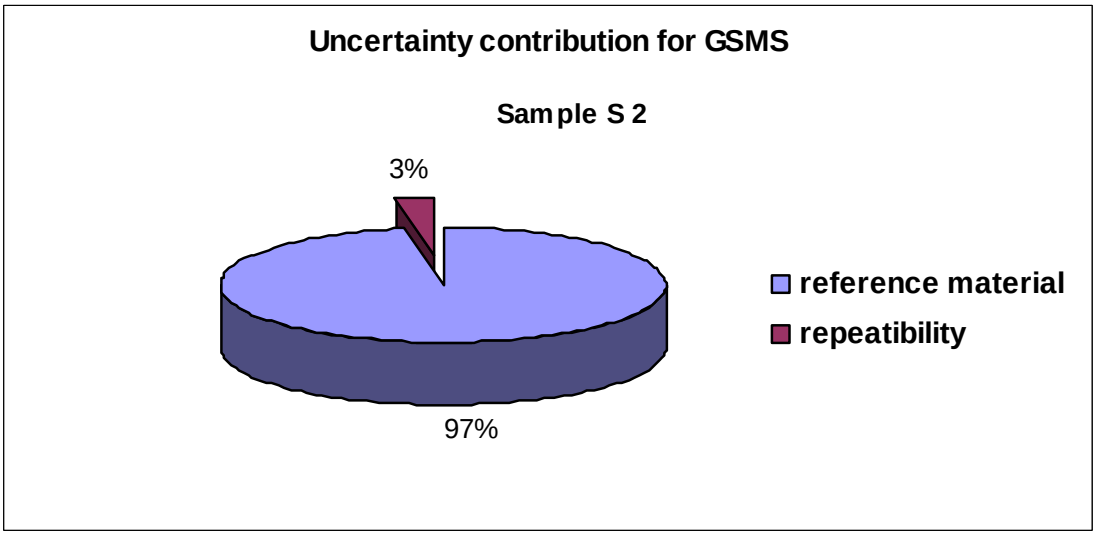

Figure 6: Uncertainty contributions for sample S 2 for GSMS

The increased relative contribution of the certified isotope reference materials to the combined uncertainty is observed when precise and accurate isotope ratios measurements are performed.

In fact very repeatable measurements must be obtained from a mass spectrometer to arrive at low uncertainty uranium isotope ratios. The repeatability of an instrument, determined as the relative standard deviation of consecutive readings, is to a large part, a measure of the stability of the ion-beams generated by ion source of the mass spectrometer and the homogeneity of the sample.

The comparison between ICP-QMS and MC-ICPMS show that although a great improvement was made with the introduction of magnetic analyser and multi-collector detectors, the plasma source flickering is still one of the main contributors to the uncertainty in ICPMS based instruments.

The results presented show that sample homogeneity in the gas form together with the electron impact process of gas source instruments still provide still the most stable mechanism of ion formation in mass spectrometry of uranium.

3.3 Advantages and disadvantages of each technique

The isotope ratio measurements executed using the ICP-QMS showed that it is the simplest technique available. The best instrumental parameters were easily determined and the measurement procedure was applied straight forwardly providing raw data that just required background and mass discrimination corrections. The high sample throughput and simple sample preparation requirements were the main advantages of this technique. The main drawback, however, was the large uncertainty values of the measured isotope ratios.

The measurements carried out by MC-ICPMS highlighted the great improvement to the measurement uncertainty brought by the incorporation of magnetic sector analysers and multi-collector detectors to plasma-based instrument systems. The measurement procedure was more complicated than that applied to the ICP-QMS and required much more attention to the memory effect correction. The sample throughput remains as a feature in this technique. Another very important advantage is the feasibility to analyse several other elements besides uranium.

The isotope ratio measurements carried out by TIMS provided results with standard uncertainties just slightly larger than the GSMS. The twenty-one filament position magazine assembled in this instrument together with computer control of the instrumental parameters enabled a very good sample throughput with high stability of the mass fractionation. This technique required a small amount of material, a very good advantage, especially when highly radioactive samples are processed although the cost per sample remains high.

The low expanded uncertainties associated with the isotope ratios determined by GSMS demonstrated that it is the preferable technique provided the sample is in the form of UF_6. Special care must always be taken to detect and remove the hydrofluoric acid (HF) usually present in the UF_6 samples and reference materials. The measurement procedure was time consuming due to the need to measure two isotope reference materials for each sample. Thus just one sample per day could be analysed. Another disadvantage of this technique is the fact that the mass spectrometer can only be used for UF_6 samples.

4. Conclusions

Isotope ratio measurements in uranium samples can be carried out accurately by the four mass spectrometry techniques investigated in this work. Nevertheless their measurement results are associated with different uncertainty levels.

The simplest technique from the operational point of view was the ICP-QMS which permitted high sample throughput but could only measure isotope ratios with expanded uncertainties in the range of 1.0 to 1.5 % for the typical uranium isotope ratios of the samples used in this study.

Although MC-ICPMS was a much more complex technique than ICP-QMS, it still permitted a good sample throughput, being able to perform isotope ratios measurements with expanded uncertainties in the range of 0.03 to 0.05 %. The correction of the memory effect was critical for accurate results.

The TIMS technique provided isotope ratios measurements with expanded uncertainties in the range of 0.03 to 0.04 %. Its main advantages were the lack of any memory effect and the small sample amount required. Like MC-ICPMS it has the great advantage of being able to analyse a broad range of elements besides uranium at very low uncertainty levels.

The GSMS was undoubtedly the technique which performed isotope ratios measurements with the smallest expanded uncertainty, in the range of 0.01 to 0.03 %. However the execution of the analysis was time consuming due to the use of the UF_6 double standard procedure applied.

The most important sources of uncertainty in all these experiments came from the reference materials used to correct the mass discrimination effect and the measurement repeatability.

As a general rule it can be said that the more precise the technique, the greater the contribution to the uncertainty from the reference materials.

5. References

[1] Benedict, M. and Pigford, T.H., Nuclear Chemical Engineering, McGraw-Hill, New York, USA, 1957.

[2] Guide to the Expression of Uncertainty in Measurement; 2nd edition; ISO-BIPM, Geneva, Switzerland, 1995.

[3] Quantifying Uncertainty in Analytical Measurement, EURACHEM-CITAC Guide, 2nd edition, London U.K., 2000.

[4] Quétel, C., Prohaska, T., Nelms, S., Diemer, J. and Taylor, P.D.P., ICP-MS Applied to Isotope Abundance Ratio Measurements: Performance Study and Development of a Method for Combining Uncertainty Contributions from Measurement Correction Factors, in G. Holland and S. Tanner (Eds.), Proceeding of the 7th Durham Conference, Plasma Source Mass Spectrometry: The New Millennium, Royal Society of Chemistry, Cambridge, 2001.

[5] Quétel, C., Vogl, J., Nelms, S., Taylor, P.D.P. and De Bièvre, P. , Comparative Performance Study of ICP Mass Spectrometers by Means of U Isotopic Measurements, Fresenius J. Anal. Chem, 368: 148-155, 2000.

[6] Richter, S and Goldberg, S.A., Improved techniques for High Accuracy Isotope Ratio Measurements of Nuclear Materials using Thermal Ionization Mass Spectrometry, International Journal of Mass Spectrometry, 229: 181-187, 2003.

[7] De Bolle, W., Muschenborn, G. and De Bièvre, P.; The Potencial of Double Collector UF_6 Mass Spectrometry for Verification of Isotopic Homogeneity; 7th Esarda Annual Symposium; Liege, Belgium; 21 May; 1985.

[8] GUM Workbench, Metrodata, Germany, http://www.metrodata.de

Destructive Analysis – Quo Vadis ?

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Abstract

The challenges in nuclear safeguards have significantly changed over the last decades. Since the early days after implementation of safeguards agreements and the Euratom regulation 3227/76, measurements of nuclear material have been the backbone of the verification measures. Non-destructive analysis was applied for measuring entire items, while samples were analysed by so-called "destructive" techniques. In this period electrochemical or physico-chemical measurement techniques (e.g. titration, coulometry or mass spectrometry) were the work horses for sample analysis. At the time, these methods provided uncertainties that were far superior to those associated with radiometric techniques. However, chemical methods require sample preparation and pre-treatment which is accompanied by the generation of waste. Over time, the waste costs have grown almost exponentially, while on the other side the performance of the radiometric techniques has steadily increased. Radiometric techniques were developed and adapted also for sample analysis, the hybrid k-edge certainly being the most prominent example.

The increasing number of samples subjected to accountancy and verification measurements led to a balanced use of chemical, physico-chemical and radiometric methods in today's analytical laboratories. The introduction strengthened safeguards, the implementation of the additional protocol and, consequently, the new sample types triggered the transfer of analytical techniques from the environmental area, from materials science to the safeguards community. Environmental analysis and nuclear forensic science are experiencing significant developments. Again classical radiochemical and physico-chemical techniques provide important information on the nature of the sample under investigation. Detection and categorization of seized material from illicit trafficking has obviously become an area where non-destructive methods are applied.

The more specific questions will be asked in safeguards with respect to a given sample, the more investigative analytical methodologies will be required and the more thorough, interpretative and comparative evaluation of results needs to be done. Specific applications, possibly in combination with only minute amounts of sample call for methods of high sensitivity, low detection limits, high selectivity and high accuracy. The selection of the method or combination of methods is done according to the sample and according to the information required.

In order to develop a versatile and flexible answer to present and future challenges in safeguards analytical measurements we need to change our viewpoint and overcome thinking in established categories like DA and NDA.

Keywords: destructive analysis, safeguards measurements

Introduction

The terminology "Destructive Analysis" (DA) and "Non-Destructive Assay" is specific to the area of nuclear safeguards. No other community dealing with material analysis (e.g. AOAC, Eurachem) applies such a distinction. Irrespective of the scientific reasoning behind this particular categorisation of measurement techniques, the terminology has been used for many years. The first edition of the IAEA Safeguards Glossary[1] there is no definition of destructive analysis provided. In contrast to that,

"Non-destructive assay" (NDA) is defined as "measurement of the nuclear material content or element or isotopic concentration of an item without producing significant physical of chemical changes in the item." A definition of Destructive Analysis is given in the second edition of the Glossary[2], specifying the "determination of nuclear material content and, if required, of the isotopic composition of chemical elements present in the sample. Destructive analysis normally involves destruction of the physical form of the sample."

It should be noted that according to these definitions, NDA refers to the measurement of "items", while DA refers to "samples". Over the years, both areas experienced significant changes in terms of instrumentation, analytical performance (i.e. accuracy and precision), costs and application. Furthermore, the recent challenges in safeguards have also affected the use of certain measurement techniques and initiated new techniques and new applications.

Traditional Areas of application

In traditional safeguards the measurement of samples is essential to the operator for establishing a quantitative and correct nuclear material accountancy and to the safeguards authorities for the independent verification of the operator's declaration. Particularly, for facilities with high material throughput accurate measurements are required in order to achieve the safeguards objective, i.e. "the timely detection of diversion of significant quantities of nuclear material". For this purpose samples are taken and shipped to a specialised laboratory for analysis. Figure 1 shows the number of samples analysed by two laboratories and the evolution over time. As can be seen there is no decrease in

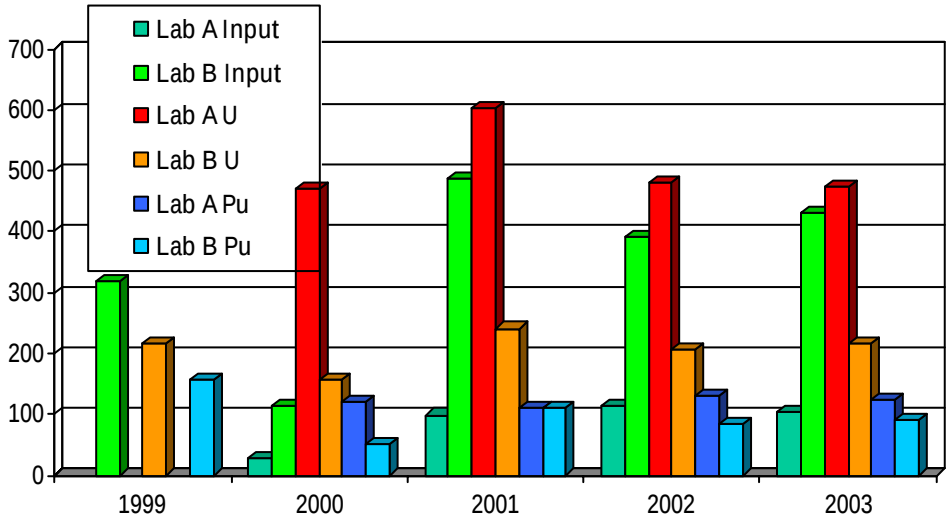

Figure 1 Number and type of samples analysed in a safeguards laboratory (A) and in an operator's laboratory (B)

samples being shipped to laboratories for analysis.
Samples arriving at a laboratory are basically analysed by electrochemical methods (titration, coulometry) and by mass spectrometry. This allows elemental and isotope assay. These methods offer a short traceability chain and high accuracy and precision.

These advantages are of key importance for the certification of reference materials. Careful measurements have to be carried out and techniques offering highest possible accuracy and precision have to be applied. In order to establish a comprehensive and transparent uncertainty budget, all uncertainty components need to be fully understood. For certification purposes one uses preferably "primary methods of measurement" as defined by the BIPM[3]. Reference materials form also an important tool for the validation of so-called "calibration free methods". Those are increasingly being used in combination with model calculations (e.g. Monte-Carlo) in order to reduce the calibration effort.

Destructive analysis is also used for the certification and/or verification of secondary reference materials. Secondary reference materials may serve for quality control or for specific calibration purposes (e.g. for K-edge instruments).

Instrumental Developments

Over the past decade we also experienced significant instrumental developments. Taking the example of thermal ionisation mass spectrometry, there was a transition from analogue electronics to digitised signal procession and instrument control. The detectors show improved signal-to-noise ratio and the signal amplifier have an increased dynamic measurement range. The possibility of simultaneous detection of ion currents at different mass positions has been complemented by new methodologies for amplifier cross calibration, thus eliminating an important source of systematic error (Type B uncertainty component). This led to an improvement of the measurement precision (repeatability) of more than one order of magnitude.

In safeguards data evaluation, the measurement uncertainty has become a minor component. The predominant contribution arises from sampling or from the bulk measurement.

Methodological Developments

In nuclear analytical laboratories radiometric techniques are increasingly used for sample analysis. The most prominent example is certainly the Hybrid K-edge instrument, which makes use of X-ray absorption of uranium for the concentration measurement and of X-ray fluorescence for the U/Pu ratio measurement. The typical relative uncertainties achieved today are around 0.2% for uranium and 0.5% for plutonium. This is sufficient for partial defect measurement. Consequently, this technique is replacing for practical purposes the primary methods of measurement (titration, coulometry, isotope dilution mass spectrometry). For bias defect detection (particularly in high throughput facilities) at least a subset of samples has to be analysed by methods with lowest possible uncertainty (isotope dilution mass spectrometry) in order to achieve the required detection probability. In combination with the large size dried spike technique, eliminating dilution errors, the uncertainty of the entire analytical process can be minimised.

The combination of radiometric and chemical methods offers increased flexibility and serves also for internal quality control purposes (via method intercomparison).

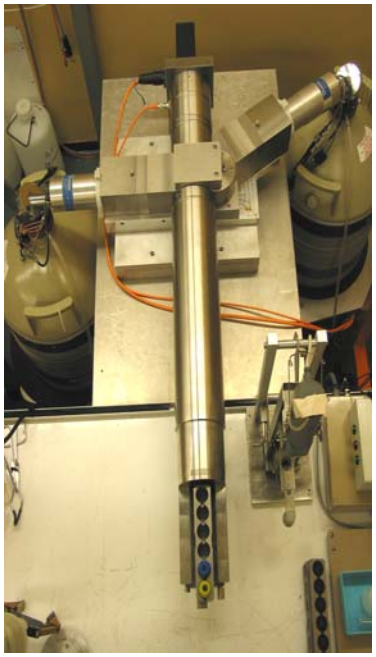

Figure 2 The Hybrid K-edge instrument is equipped with a sample changer that enables high sample throughput

New Areas of Application

The changes in the safeguards world (implementation of the Additional Protocol, environmental sampling as part of routine inspections, etc.) an increasing variety of samples is being subjected to analysis. Particularly **samples from complementary access** show a wide range of matrices and concentration in analyte. Also the analytical requests are more specific and more tailored to the particular inspection. The question of verifying the absence of undeclared activities is the key point in these investigations. In contrast to traditional verification samples, we know very little about such samples from complementary access. The composition of the matrix may be unknown, the concentration or even the presence of uranium may be unknown. These unknowns obviously pose a challenge to the analyst. Consequently, the analytical approach needs to be adapted to each individual sample and a flexible methodology has to be developed.

Additionally, samples of nuclear material that were intercepted during illicit trafficking (see example shown in figure 3) need to be analysed. These **nuclear forensic investigations** go far beyond the traditional safeguards analysis. Here the primary questions are on the nature of the material, its intended use and its origin. The chemical composition, the isotopic composition (of major and minor constituents), the macroscopic parameters, the microstructure, chemical impurities and the age of the material are relevant information that may provide clues on the origin of the material. Again, the scope of the analysis is much wider than in traditional safeguards and the interpretation of the measurement data is more complex. The first investigations are usually performed using NDA techniques (e.g. gamma spectroscopy of the material without unpacking), more detailed analysis using chemical and material science techniques are preformed after unpacking and visual inspection.

Figure 3 Example a seized nuclear material wrapped in lead foil

The area of nuclear forensic science is relatively young and considerable research and development work is going on in this area. On the one hand we still need to find out which parameters are characteristic for a certain process or facility. Apart from isotopic patterns, also elemental patterns (i.e. the chemical impurities in a nuclear material) may provide useful information on the origin of the material. On the other hand the compilation of reference data is important in order to facilitate source attribution.

A completely new challenge consists is the preservation of classical forensic evidence from nuclear material or its associated materials (packing, shielding). Fingerprints, hair, fibre, DNA that are associated with the seized material may provide useful information on the criminal suspect involved in the case. Such investigations require appropriate facilities that satisfy radiation protection needs (e.g. glove boxes) and offer at the same time a clean environment for sample taking.

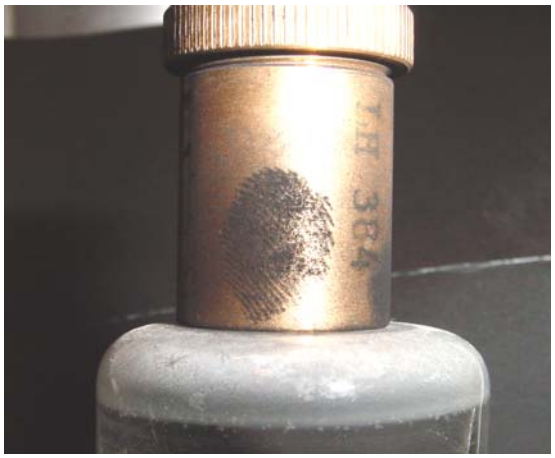

Figure 4 Fingerprint taken inside a glove-box from a plutonium contaminated brass container.

Environmental measurements for safeguards have been introduced with the IAEA's "93+2 programme". The majority of these samples presently consist of swipe samples. These swipes are analysed for uranium particles using mass spectrometric techniques. Secondary ion mass spectrometry is applied for isotope analysis of micrometer sized particles. Figure 5 displays the spatial distribution of the two main uranium isotopes in a sample, clearly identifying four particles which mainly consist of ^{238}U. The increasing number of swipe samples to be analysed obviously calls for efficient screening techniques and for automation of the particle search and particle analysis routines. Figure 6 clearly indicates the growing amount of requests for swipe sample analysis of the past five years.

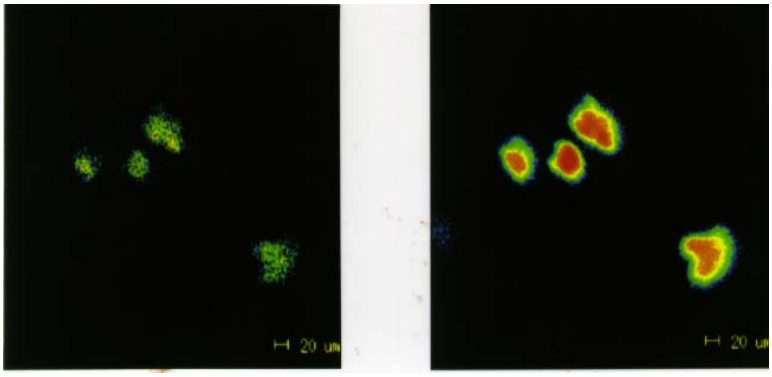

Figure 5 Visualisation of the ^{235}U (picture on left side) and ^{238}U (right side) abundance in four particles

Furthermore, there is a need for particle reference materials with certified isotopic composition and with defined particle size. These materials serve for instrument calibration, for the determination of the ionisation efficiency and for quality control. Since the sample preparation (i.e. the transfer of particles from the swipe to the sample holder for the SIMS) is an essential step, also reference swipes containing a known amount of particles appear to be very useful.

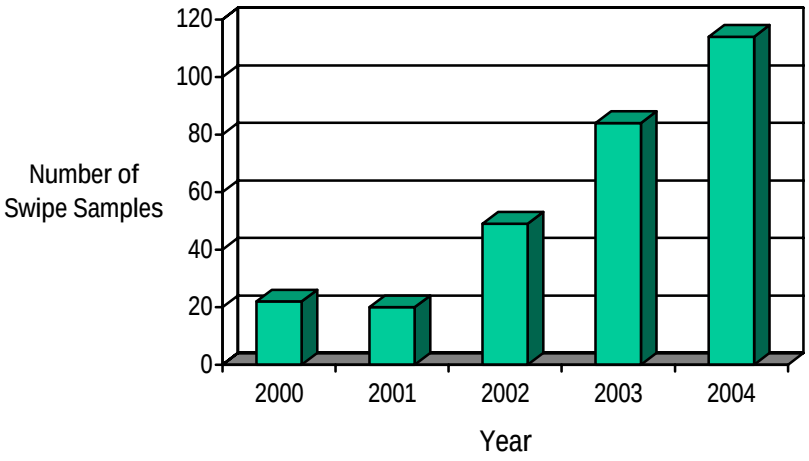

Figure 6 Amount of swipe samples analysed at ITU for various customers over the past five years.

Conclusion

We note that the number of samples of nuclear material analysed in safeguards analytical laboratories and in operators' laboratories has remained practically at constant level of past couple of years. The analytical methods which are applied for sample analysis are based on a balanced use of radiometric, chemical and mass spectrometric techniques. This is mainly due to the high degree of maturity which the radiometric techniques have achieved.

In addition to this more traditional application, we also note an increasing number of "exotic" samples from complementary access and from non-safeguards activities. These samples require a more investigative (sometimes even a detective-like) type of analytical work. The materials to be investigate range from picogram amounts in particles or environmental samples to gram amounts in traditional safeguards samples. New types of matrices and a growing list of analytes challenge the analyst. The methods have to be adapted to the individual samples and a larger variety of parameters have to be investigated. This can only be achieved by using a suite of complementary measurement methods, irrespective whether they would be categorised as being "DA" or "NDA". The current challenges in sample analysis take us beyond these traditional categories.

These considerations lead to the question whether the existing mechanisms and structures within ESARDA, particularly in the discipline oriented working groups, are still appropriate and allow properly addressing the current and future challenges associated with sample analysis. We should wonder whether the categories DA and NDA do still reflect the reality in our laboratories. The increasing complexity in sample analysis requires a multidisciplinary approach in order to provide the most useful answers in the most efficient way.

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[1] IAEA Safeguards Glossary, IAEA/SG/INF/1 (Rev.1), 1987 Edition, International Atomic Energy Agency, Vienna, 1987

[2] IAEA Safeguards GLossary, International Nuclear Verification Series No.3, 2001 Edition, International Atomic Energy Agency, Vienna, 2001

[3] M. J. T. Milton, T.J. Quinn; Primary methods for the measurement of amount of substance; **38**, 289-296 *Metrologia* 2001

Session 4

National and Regional Systems

Aspects of Safeguards Application in Lithuania after Accession to the European Union

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Abstract:

On May 1 2004 Lithuania together with 9 more countries became a member of the European Union (EU) and at the same time of the European Atomic Energy Community (EURATOM). From the nuclear safeguards application in Lithuania point of you that meant the following changes: start of the Euratom safeguards application and shift from the bilateral safeguards agreement with the International Atomic Energy Agency (IAEA) to the trilateral one between the EU non-nuclear weapon states, Euratom and the IAEA.

Previous EU enlargement in 1995 involved similar changes related to safeguards application. It might suggest that 2004 enlargement was not something new as far as the safeguards concerned. However, there were certain changes in the safeguards field within the European Commission prior to 2004 enlargement. Considerable progress in the IAEA strengthened safeguards implementation was achieved in Lithuania until the membership in the EU too.

Influence of varied circumstances in the European regional safeguards system to the safeguards application in Lithuania and the shift from the bilateral safeguards agreement to the trilateral one will be the focus of this paper.

Keywords: experience; state system; additional protocol

1. Introduction

Lithuania declared its independence in March 1990 and in August 1991 after the formal collapse of the Soviet Union the Ignalina NPP came under the authority of the Republic of Lithuania. The inherited nuclear power plant became the only facility of the nuclear fuel cycle in the country. However, the Ignalina NPP containing two units with RBMK type reactors was the outstanding installation bringing many new issues to be addressed by the young state. One of them was safeguards.

Back in early 1990s there was no experience in the safeguards application field from Lithuanian side. Safeguards as the international verification regime was not exercised in Lithuania during the soviet times. Therefore, developing national infrastructure including legal and administrative issues was evolving process, which took several years. Safeguards implementation at the facility level had to be started from scratch too. The situation was even more peculiar because of the fact that the International Atomic Energy Agency (IAEA) had no previous experience in implementing safeguards at the nuclear power plant with the RBMK type reactors. However, during the following years the Ignalina NPP, considered as the black box in the beginning, was transformed by the effort of the IAEA inspectors, national authority and the facility personnel into the well-safeguarded and transparent installation.

Changes on the international safeguards arena with the appearance of strengthened safeguards had the influence on the safeguards application in Lithuania as our country was among the first ones to sign the Additional Protocol and bring it into force. Preparations and implementation of the Additional Protocol provisions was a next stage of safeguards application in Lithuania.

The big achievement for Lithuania became a membership in the European Union (EU). However, that meant certain changes in the nuclear sector. Lithuania was obliged to shut down the Ignalina NPP and the time schedule was set. Membership in the EU indispensably affected the safeguards matters as well. Lithuania joining the EU adhered to the European Atomic Energy Community (Euratom) Treaty. The IAEA safeguards along with the Euratom safeguards and interconnection between them has become the new features of safeguards application in Lithuania.

2. IAEA Safeguards in Lithuania

The cornerstones of safeguards application in Lithuania were signing of a comprehensive safeguards agreement and its additional protocol. Agreement on the Safeguards Application between the Government of the Republic of Lithuania and the IAEA was signed on 15 October 1992. The Additional Protocol was signed on 11 March 1998, ratified two years later and finally came into force on 5 July 2000.

Prior to the implementation of the Additional Protocol in Lithuania, the new process in the nuclear fuel flow was started at the Ignalina NPP. The dry spent nuclear fuel storage was commissioned next to the power plant. It came into operation in 1999 and spent nuclear fuel transfers from the units to the dry storage began. That substantially increased the IAEA inspection activities in Lithuania. Agency inspectors and technicians worked 90 days in Lithuania in 1998 and in 1999 this number doubled. [1] It went even higher in subsequent years. [2] Transfers of the spent fuel gave an impulse to update technical measures used for safeguard purposes at the Ignalina NPP. Data collection and surveillance systems were upgraded.

In the beginning of 2001 Lithuania submitted the initial declaration pursuant to the Additional Protocol requirements. In 2001, 2002 and 2003 the IAEA carried out four complimentary accesses: three at the Ignalina NPP site and one at the Institute of Physics. In the Safeguards Implementation Report (SIR) for 2003 the IAEA stated that for 19 States the Agency found no indication of diversion of nuclear material or of undeclared nuclear material or activities. [3] Lithuania was among those 19 States and the positive conclusion for our country was drawn for the first time.

In 2004 the IAEA continued its activities in implementing the Additional Protocol and three more complementary accesses were carried out in Lithuania. Although the SIR for 2004 has not been issued yet, we expect that the Agency will be able to reiterate the conclusion for Lithuania and for 2004.

The positive conclusion paves the way for the integrated safeguards implementation in the State. Integrated safeguards approach as the optimum combination of all safeguards measures available to the Agency under comprehensive safeguards agreement and additional protocol is developed by the IAEA for the state as a whole. Integrated safeguards approaches are developed for specific facility types as well. The Ignalina NPP with on-load fuelled RBMK reactors is the only such nuclear power plant in operation under safeguards all over the world. The other are in Russian Federation, a nuclear weapon state, and the Chernobyl NPP in Ukraine is closed down. Therefore, the focus of the IAEA in developing the integrated safeguards approach for on-load reactors is on CANDU-type reactors. However, the general principles such as performing a short notice or unannounced inspections, provision of near-real time information on nuclear material flow to the IAEA are applicable in any case. Discussion between the IAEA, State and operator on specific implementation of these principles has already begun and expectations are that at the end of 2005 the integrated safeguards will start to be implemented in Lithuania.

The current situation when the first unit of Ignalina NPP has been closed down and new projects and processes related to nuclear fuel use and storage are foreseen is not very much appreciative. It would be easier to develop and implement the approach for a stable and steady situation. However, the transfer of not fully burned fuel from unit one for reuse in unit two will take place. This is completely new activity over power plant's operation time. The existing dry storage has been almost filled in and the transfers of spent fuel will cease for some time. However, the new dry storage will be commissioned in a few years time and the transfers will be resumed. The integrated safeguards approach has to embrace all current and future activities, as the aim is to have the approach applicable for a long term.

3. Euratom Safeguards in Lithuania

Lithuania became a member of the EU on 1 May 2004. Treaty of Accession of the Czech Republic, Estonia, Cyprus, Latvia, Lithuania, Hungary, Malta, Poland, Slovenia and Slovakia signed in Athens on 16 April 2003 contained Protocol No. 4 on the Ignalina nuclear power plant in Lithuania. Article 1 of the protocol stated that Lithuania commits to the closure of unit 1 of the Ignalina NPP before 2005 and of unit 2 of this plant by 31 December 2009 at the latest. [4] Lithuania has already fulfilled the first part of the commitment by closing down unit 1 on 31 December 2004.

Upon accession to the EU, the provisions of the EURATOM Treaty as part of the "Acquis Communautaire" have come into force in Lithuania. To prepare for fulfilment of the obligations under Chapter VII of this Treaty, the European Safeguards Office launched the ACCESS (Applicant Countries Co-operation with the Euratom Safeguards System) project in 2000. The purpose of the project was to help the would-be EU countries possessing nuclear materials and operators of nuclear facilities to change over to a new procedure of nuclear material accounting. Working directly with the IAEA, reports would be prepared in line with the Agency requirements and submitted directly by the Lithuanian national authority to the IAEA. While in the EU, nuclear material holders and nuclear facility operators send nuclear material accounting reports to Luxembourg and the reports are send to the IAEA from there. The change of information flow had its own implications and the ACCESS Steering Committee meetings provided a good forum for discussions, although, the focus was on the ACCESS software development, which would facilitate implementation of the EU nuclear material accounting rules.

Following a High Level Experts Group (HLEG) report, the former Euratom Safeguars Office was fully integrated within the Commission's Directorate-General for Energy and Transport (DG TREN). [5] The administrative changes apparently did not serve well for the successful course of the ACCESS project. The project was not completed by 1 May 2004 and it is not yet successfully realized one year after. Another of recommendations made by the HLEG was a review of Regulation on application of Euratom safeguards. A process of adoption of a new regulation protracted and it was not officially in force on the day of the EU enlargement.

During the bilateral meeting in Vilnius in December 2003, DG TREN Directorate H and I representatives and Lithuanian counterparts discussed the road map for the steps, which need to be taken for the successful Lithuania's resettlement in the Euratom safeguards system. However, pending completion of the ACCESS project as well as coming into effect of the new Regulation were the circumstances, which Lithuania as the new EU member was confronted with in the Euratom safeguards field. The DG TREN sent the letter in the beginning of May 2004, in which it requested the copies of nuclear material accounting reports prepared for the IAEA to be sent to the Commission Headquarters in Luxembourg as the new Euratom requirements for accounting were not in force and actually could not be fulfilled without the ACCESS software. Parallel reporting to the IAEA continued as usual.

The first DG TREN Directorate I inspectors' visit to Lithuania occurred yet prior to the accession. During the IAEA inspection in March 2004 at the Ignalina NPP, Euratom attended as observers. The independent inspection by DG TREN Directorate I took place already at the end of May 2004. Euratom inspectors came to verify the basic technical characteristics provided by the Ignalina NPP. The subsequent Euratom inspection of verification of the initial inventory was carried out together with the scheduled IAEA inspection. Euratom inspectors were present during the IAEA annual physical inventory inspection at the Ignalina in 2005. According to guidelines on the frequency of the inspections at nuclear reactors presented by the Commission in a new approach of Euratom safeguards implementation, that promises to be the only inspection under the Euratom Treaty in Lithuania in 2005.

The new approach of Euratom safeguards implementation developed by the Commission, as completion of internal review since the HLEG report, triggered a lot of debate among the parties involved. Although it presents drastic changes in what had developed in the safeguards application field within the EU, it does not substantially affect Lithuania. Whatever Euratom approach is, it is a new one for a newcomer in the EU. However, the Euratom safeguards are entangled with the IAEA safeguards. Interconnections and information flow between operator, national authority, Commission and the IAEA can hardly be separated just for the purposes of one safeguards system or other.

4. IAEA and Euratom safeguards conjunction

After Lithuania joined the EU, it became necessary to suspend the bilateral agreement with the IAEA on application of safeguards and its additional protocol, and adhere to relevant documents between the IAEA, Euratom, and the EU non-nuclear weapon states. This was not a quite new situation as it happened during the previous EU enlargements and the last time in 1995 when Austria, Finland and Sweden joined in. The only and main difference was the Additional Protocol. The strengthened safeguards system and its tool, the Additional Protocol, was under development then yet.

Lithuania already had four years of experience in implementing the Additional Protocol prior to joining the EU. Additional Protocol in the EU came into force at the end of April 2004 just before the enlargement. The IAEA took a position that changing from the bilateral additional protocol to the trilateral would not constitute a step back. Evaluation and drawing of conclusion under the additional protocol is done for states individually. Therefore, there would be no meaning to start everything from the very beginning especially as the positive conclusion for Lithuania was drawn for 2003 and move to the integrated safeguards advanced.

The clear situation with the additional protocol implementation has been one of prerequisites for a transition from the bilateral arrangement to the trilateral one. Responsibility for the European Commission and the state is actually clearly stated in the text of the trilateral additional protocol and implementation of it is not the most difficult issue for the successful transition.

More complicated issue is an inspection regime under the merged IAEA and Euratom safeguards system. New Partnership Approach (NPA) set between the IAEA and Euratom provided a clear arrangement for common inspections. However, Commission's new approach of Euratom safeguards implementation stirred up the picture. On the other hand, the Commission's way of implementing Euratom safeguards has little influence on the IAEA safeguards implementation in Lithuania. Commission's inspection regime in the framework of the new approach of Euratom safeguards implementation even is more coherent with the Agency's integrated safeguards. However, a digression from the NPA is imminent.

The last but not least prerequisite is an established and operational nuclear material accountancy reporting. Actually, this is the essential issue without which implementation of the trilateral arrangement is impossible. Unfortunately, the ACCESS project results did allow the transition to take place in 2004.

5. Conclusions

Starting from 1 May 2004, Euratom safeguards applied by the European Commission complemented the IAEA safeguards in Lithuania. Interconnection between the two safeguards systems stipulated in the trilateral arrangement between the IAEA, Euratom and the EU non-nuclear weapon states has not come into effect in Lithuania or other new EU member state during the first year of membership. The prolonged transition has been basically caused by the changes in the Euratom safeguards application field. However, that has no influence for the implementation of the IAEA safeguards in Lithuania and the integrated safeguards approach should be implemented in the nearest future.

6. References

[1] VATESI; *Nuclear Energy in Lithuania: Nuclear Safety; Annual Report 2000;* Vilnius; 2000; p 33.

[2] VATESI; *Nuclear Energy in Lithuania: Nuclear Safety; Annual Report 2001;* Vilnius; 2001; p. 22.

[3] IAEA; *The Safeguards Implementation Report for 2003;* GOV/2004/27; 2004; p. 5.

[4] http://europa.eu.int/comm/enlargement/negotiations/treaty_of_accession_2003/

[5] http://europa.eu.int/comm/energy/nuclear/safeguards/doc/high_level_group_report_en.pdf

French approach to nuclear material inventories in an 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 and at operator and competent authority level respectively, nine exercises with increasing complexity have already been carried out. These exercises have been used to validate the methodology as well as the composition of the various crisis centres and the relations with the different units involved in carrying out an inventory in an emergency.
After a short overview of French domestic regulations, this paper describes various aspects of nuclear material inventory exercises in an emergency, the methodology used and the lesson learnt from these simulations.

Keywords: inventory, crisis, exercise, lesson learnt

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 a law that has recently been replaced by 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 are 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 suggested to the competent authority by IRSN based on the specific features of the facilities and the objectives of the exercise. Different facilities are chosen in order to test the maximum of operators holding sensitive nuclear materials (plutonium and enriched uranium).

Secondly, IRSN sets up a scenario. It is assumed that a certain quantity of nuclear material, whose chemical forms may be specified, has potentially been stolen from a facility.

The practical details, such as the expected objectives of the exercise or its duration, are discussed by the common working group. The operations of physical inventory must comply with the operating rules of the facility. 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

In the event 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 level.

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.

The organisation set up by the operator is very similar to the one designed for nuclear safety matters. 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 request;
- 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 or 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.

6. Lesson learnt

Nine exercises have been organised since 1993 in various types of facilities (research reactor, uranium metal processing workshop, research laboratory, reprocessing plant, etc.). Until now, only facilities holding sensitive nuclear materials (plutonium, uranium enriched with 20% uranium-235 or more and uranium-233) have been concerned by these exercises.

The complexity of exercises has increased in order to become as close as possible to actual emergency conditions. For example, we have already tested the following configurations successively: one facility involved in the exercise, several facilities located on the same nuclear site, several facilities located on different sites, several sites under the responsibility of different nuclear operators and even two sites and the transport of nuclear materials between them. In addition, simulated media pressure was introduced into the last two exercises.

For the authority and its technical support these exercises show:

- the need to clarify internal organisation and links with other emergency situations (safety emergency cases);
- the need for expert training, in particular to reply quickly and clearly to media requests;
- the importance of IRSN holding basic data about each site and facilities, both on physical protection as well as the audited nuclear materials. These data may be used to acquire instant knowledge of the main site characteristics, the activities of the facilities, the quantities of nuclear materials, the number of articles held (or at least an order of the magnitude of this number) and the resources available for classified data transfers;
- the need for the authority to receive regular, frequent information from the facilities involved.

These exercises also highlight for the operators:

- the importance of using seals, after characterisation of nuclear materials, to facilitate and accelerate physical inventory taking;
- the need to plan action sequences to improve the efficiency and reduce the time for the inventory taking;
- the need to define solid methodology and type of non-destructive measurements to be implemented in an emergency inventory.

For the entire emergency organisation, feedback from the exercises suggests:

- the need to clarify missions and relations between the various entities involved in this type of situation;
- the need to define and implement specific resources, in particular for classified data transfers;
- the need to prepare reflex sheets, adequate procedures and up-to-date lists of telephone and fax numbers.

7. Conclusion and perspectives

This initiative has allowed a methodology to be established to prepare and implement nuclear material inventory exercises in an emergency. The increased complexity of the exercises has resulted in extensive preparation of the nuclear operators as well as an efficient organisation of the authority and its technical support body for a real emergency.

In addition, these exercises have been used to define the necessary documents, procedures and communication resources in the event of an emergency for a variety of facilities. All deficiencies or inaccuracies found in the documentation during the first exercises have been corrected.

Finally, these exercises have made the operators involved, the authority and its technical support aware of the importance of efficient training for this kind of event.

The challenge for the next exercises is to work on a more realistic scenario, by:

- increasing the simulated media pressure;
- considering a broader scope of potential theft, perhaps in connection with a malicious act or blackmail by terrorists;
- linking the nuclear material crisis with a safety emergency case.

AUDITING FOR QUALITY IN NUCLEAR SAFEGUARDS: THE EUROPEAN COMMISSION'S PERSPECTIVE

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Abstract

The European Commission embarked in 2004 on a review of its policy on nuclear safeguards. The role of the operator of a nuclear installation as the first responsible for nuclear material accounting and control has been confirmed and clarification has been provided on the main elements of the Commission's verification approach.

While independent controls of nuclear material are maintained, verifications will concentrate on the performance of nuclear operators and their capability to account for and to control quantities of nuclear material in their possession at any location and at any time. Audit methods will play an important role in this assessment

This paper describes the elements of this audit approach as applied in three levels: compliance control, performance control and credibility control. Compliance refers to the commitments of nuclear operators emanating either directly from the Euratom Treaty, or from the Euratom Safeguards regulation or from derived legislation. Performance control addresses the capability of operators to establish, implement and maintain high quality nuclear material management systems. Finally, credibility control deals with the need for Commission inspectors to perform independent physical verifications on nuclear material.

Keywords: policy; audit; performance; quality

1. Introduction

It is the role of the Commission to implement Chapter VII - Safeguards - of the European Treaty. In its Article 77, the Treaty states that the Commission shall satisfy itself that: nuclear materials are not diverted from their intended uses as declared by the users (Article 77(a)) and the obligations assumed by the Community under agreements concluded with third States or international organisations are complied with (Article 77(b)).

The Commission implements these obligations on the basis of a number of legal tools:

Articles 78 and 79 of the Euratom Treaty put obligations on the nuclear operators to have nuclear material accounting systems in place and to report regularly to the Commission. Articles 81 to 83 give the right to the Commission to send inspectors and to impose sanctions in case of infringement of the EU legislation.

Commission Regulation (Euratom) No. 302/05 on the application of Euratom safeguards lays down detailed requirements for the operators' information and reports as regards technical characteristics and nuclear material accounting.

Particular Safeguards Provisions (PSP) are individual decisions of the Commission and part of the legal acts of the EU legislation. They are provided for in the Regulation 302/05 under Articles 7, 8 and

35 and set out, *inter alia*, the operator's obligations with regard to the structure of records and reports related to the nuclear material accounting measures.

In applying its nuclear safeguards policy, the Commission deals directly with nuclear operators. This direct relationship is quite different from the position of the IAEA where conclusions regarding the absence of diversion of nuclear materials are made on the level of an entire state. The Commission's approach to nuclear safeguards institutes the nuclear operator as the first person responsible for establishing, implementing and maintaining adequate control of nuclear materials within his installation. The Commission's services intervene as a second control layer to conclude on the quality of the implemented operator procedures.

The quality of the operator's control systems will essentially lie in their ability to: a) maintain adequate knowledge of the declared nuclear material; and b) quickly detect with high likelihood the loss of a small quantity of nuclear material.

The safeguards approach of the Commission is based on an efficient audit of nuclear material accounting and control systems, with appropriate random checks of procedures, records and of the physical reality. The Commission continues its independent and direct inspections (physically accessing nuclear material) including unannounced, short-notice or random inspections aimed at improving the effectiveness of verification. An optimal balance between headquarter activities and on-site inspections will be pursued.

2. Three components of nuclear material control

Verifications by the nuclear safeguards inspectors of the Commission address the following interactions (figure 1):

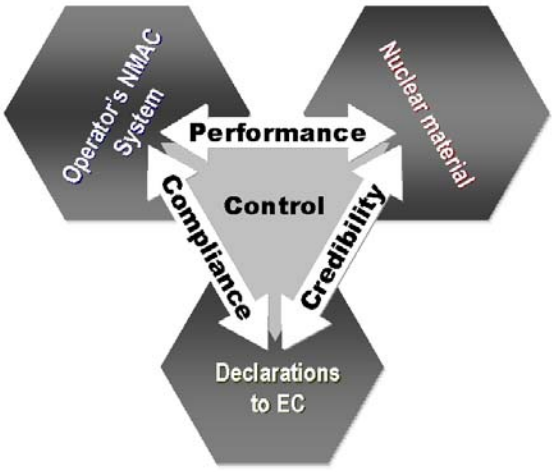

Figure 1: Three type of nuclear material control

1. Compliance control: The declarations made by the users (operators) to the Commission vs. the legal requirements and the operator's nuclear material accounting system.

2. Performance control: The accounting and control elements of the operator's nuclear material management system vs. the way the nuclear material is received, processed, stored and shipped.

3. Credibility control: The quantities of nuclear material contained in the accounting declarations to the Commission vs. the physical reality (as verified during inspections).

2.1. Compliance Control

Within the scope of compliance control, the Commission services will verify the operator's declarations concerning the technical characteristics of the installation, the operating records for the accounting of nuclear material and the organisation of the nuclear material accounting and control system. This control involves a double comparison:

- the declarations' structure against the formal legal requirements (The Euratom Treaty, the Regulation 302/05 and the Particular Safeguards Provisions) and

- the declarations' content against the facts (by examining and validating the operating records and technical data).

The Euratom Treaty establishes in Article 78 that "anyone setting up or operating an installation shall declare its Basic Technical Characteristics (BTC)". Article 79 provides the basis for the Regulation 302/05 to define the extent of this declaration. This declaration is prepared by the operator, as specified by Article 3 of the Regulation, on the basis of a questionnaire per type of installation.

The Regulation 302/05 foresees in its Article 6 that the Commission decides on Particular Safeguards Provisions (PSP) to be drawn up in order to specify the "procedures by which the persons or undertakings concerned shall meet the requirements in relation to safeguards imposed on them".

This Commission decision imposes on nuclear operators a minimum set of rules for the implementation of a Nuclear Material Accountancy and Control (NMAC) system providing a structure of records, reports and procedures related to the nuclear material accounting measures. These include among others the definition of Material Balance Areas (MBA) and Key Measurement Points (KMP) for the determination and reporting of flows and inventories of nuclear material, as well as the structure and periodicity of Inventory Change Reports (ICR), Physical Inventory Listings (PIL) and Material Balance Reports (MBR).

Compliance control includes both Headquarter and On-site activities. Headquarter activities are primarily the control of consistency and conformity of the operator's declaration and reports with requirements. On-site activities relate to conformity checks of the operator's declarations with reality, in particular the comparison of declarations with operating records and the control that declared technical characteristics of the installation (including the use of nuclear materials) are matching reality.

2.2. Performance Control

The aim of performance control is to evaluate the operator's NMAC system on strengths and weaknesses and to verify that the system is capable of detecting diversion of small quantities with high likelihood in a short amount of time. Audit methods are part of the Commission's performance control toolbox. Hereby the objective is to obtain assurance that the operator's nuclear material management system performs effectively to high quality levels.

The audit methods applied check if the NMAC system is:

- described with enough detail,
- effectively implemented and operational,
- of high quality and robust,
- capable of achieving the envisaged diversion risk control,

A NMAC model serves as reference for these audits. This model includes essential sections on management structure, nuclear material accounting and measurement programmes along with their independent internal controls.

Figure 2 presents the structure of a typical NMAC system model:

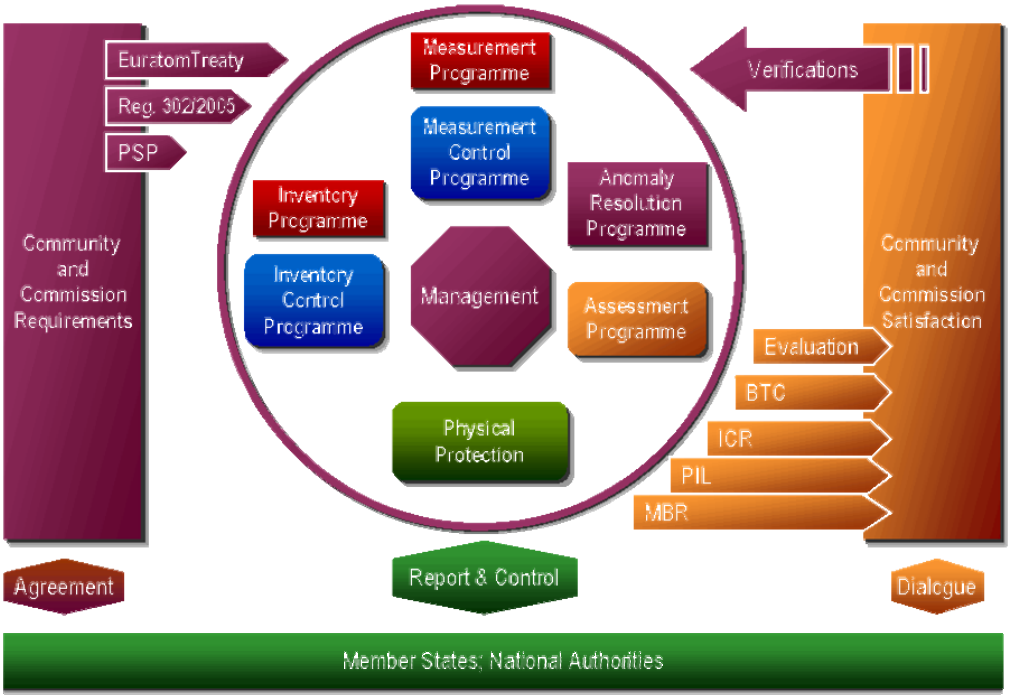

Figure 2: Structure of a typical NMAC system model

The following description of the assessment of some basic NMAC system elements is provided as an example of the guidelines along which inspectors are proceeding when applying audit techniques:

1. The nuclear material management structure of the operator must exhibit clear and distinct responsibilities for all NMAC key functions and be independent from the production department. The key responsibilities and the respective qualifications need to be well defined. The management must be committed to promote and ensure the approval and use of documented NMAC processes as well as an adequate internal review policy.

2. A measurement programme is set up for the determination of all relevant quantities and data, establishing and maintaining documented procedures. This programme is also applied, where appropriate for the acquisition, preparation and use of reference standards and for calibration of all measurement systems.

3. In addition, a measurement control programme is set up completely separate from the measurement programme and assigned to a key responsible having independence of action as well as adequate authority to obtain all the information required to monitor and evaluate measurement quality.

4. An inventory control programme is established, not only for regularly performing complete physical inventories and reconciling, resolving and reporting on found differences but also in order to maintain knowledge on an ongoing basis about the location, quantities and composition of nuclear material . The inventory control programmes ensures that storage, handling and measurement methods are adapted to the diversion risk control goals and maintains methods and procedures for early detection, investigation and resolution or recovery of missing material.

5. Next to an anomaly resolution programme, the NMAC system model includes a programme for the independent assessment of the effectiveness of the operator's accounting and control. Assessment results are documented and reported to the plant management and corrective actions are pursued.

In this performance control concept, audit criteria are defined according to this NMAC model. The inspector-auditor determines to which extent the audit criteria are fulfilled and establishes a installation performance rating. This rating may be used for ranking comparable installations and for the determination of the amount of inspection effort to be invested in follow-up control, modulating the range and depth of additional audit activities as well as independent physical verifications.

Performance control audits are considered as an important tool helping to improve the quality of the operator's control system. By identifying facts displaying shortcomings, the audit serves to bring about the awareness of the operator, the identification of the root causes and the way to resolve the problem in order to avoid reoccurrence.

2.3. Credibility Control

A set of tasks to supplement the audit activities will include the verification by the Commission's services that the operator himself does not divert nuclear material from its intended use. This risk, although it is considered to be extremely low, may not be adequately covered by auditing the operator's NMAC system without accessing the nuclear material for physical verification. During these inspections, random physical checks at an appropriate level of intrusiveness will be applied either by using Commission instrumentation or by making best use of operator measurement systems, data and samples.

2.4. Combination of different types of control

During on-site inspections, verification activities belonging to the three types of control described before are combined. Sampling techniques are used systematically but when sample sizes are normally initially low, findings may call for a deepening or widening of the investigations as represented in figure 3.

All parts of the installations and all elements of the operator's NMAC system can be selected for verification. The Commission will independently set the frequency and the degree of intrusiveness of the inspections as a function of the risk perceived for a given operator including the results of audit activities and previous verifications.

The inspection activities include physical verifications by Commission inspectors of the inventories and flows of nuclear material based on a random selection of a small number of items. These physical verifications may serve the purpose of compliance control (e.g. verification of the declared operator measurement capability), of performance control (verification that the implementation of established procedures is respected) or of credibility control (examining credible diversion scenarios).

On-site verifications may mean the carrying out of no-notice, short-notice, or random inspections when appropriate. For installations where information exists or previous inspection results indicate that there might be an increased suspicion for deliberate falsification of the data or for diversion of nuclear material, the number of verified items may be increased as required.

In applying this risk-informed and performance-based approach, the Commission services may also decide, on a case by case basis, to apply containment and surveillance measures and/or unattended measurement systems in order to "freeze" nuclear material or to provide for enhanced or long-term observation of plant activities or nuclear material flows, where judged necessary.

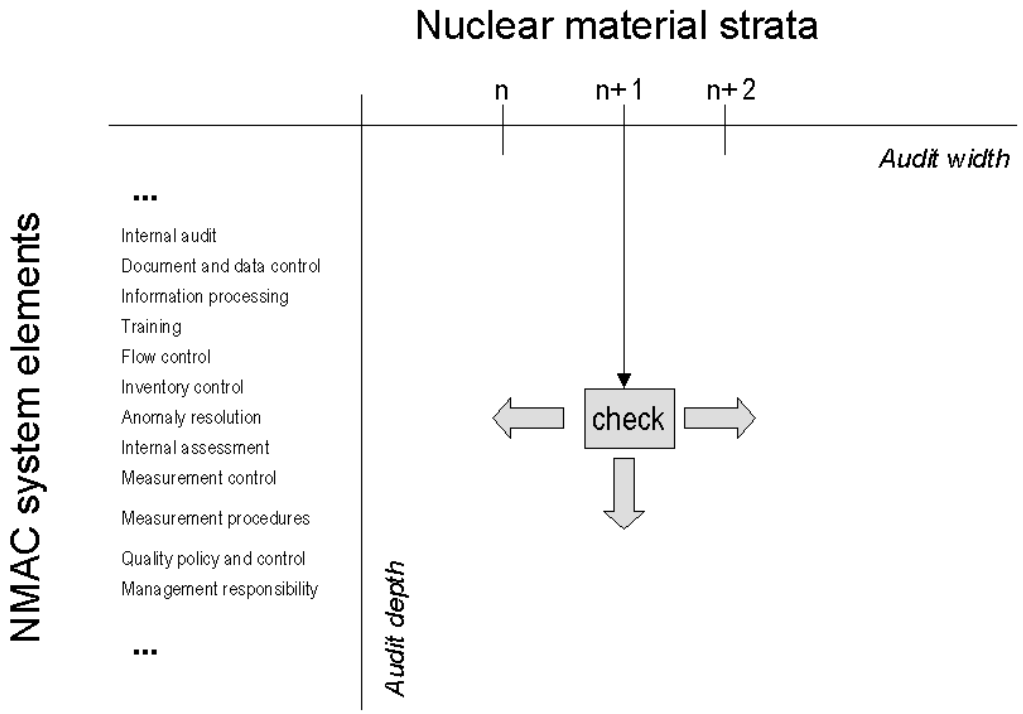

Figure 3: Widening and deepening of verifications

3. Conclusion

The described perspective of the European Commission is a major improvement in the field of nuclear safeguards. The primary objective of a credible safeguards system must be the protection of the European citizen. The Commission emphasises in its nuclear safeguards policy the basic role and responsibility of nuclear installation operators and aims through its verifications at maintaining a high quality level of nuclear material control.

The European Commission is endowed, by the Euratom Treaty, with enforcement mechanisms including sanctions in case of non-respect of the obligations. This provides a supranational authority that is second to none to any national and international organisation in charge of implementing a safeguards regime.

However, Euratom nuclear safeguards is a concept more limited than nuclear security which has nowadays to be considered in the European Union as a partnership. On the one hand, chapter VII of the Euratom Treaty covers only some materials (plutonium, uranium and thorium) and on the other, responsibilities for nuclear security are spread over several actors: individual companies, the Member States, the Commission and the International Atomic Energy Agency. Only such a partnership, with a clear share of responsibilities, provides the best guarantee to the European citizen.

IAEA/ABACC Procedures for the Joint Auditing of Accounting Records

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Abstract:

The Procedure for Joint Auditing of Records is the result of a series of discussions that started four years ago. In November 2000, ABACC and IAEA met for the first time aiming at the implementation of the joint system for auditing accounting records during inspections. A comparison study of the auditing procedures used by each of the Agencies up until that time was carried out. The study concluded that the main differences in the procedures were in how the ending date for the auditing period was defined and in the working papers used. The Agencies decided that modifications were to be made in ABACC's Software for the Auditing of Records (SARA) by adopting the IAEA criteria for the ending date for the auditing period and generating information in the format required by the IAEA.

One year later, at a meeting held in Vienna, ABACC introduced the new Software for Joint Auditing of Records (SJAR). SJAR was tested by the IAEA through simulation exercises based on actual data from representative inspections performed in Brazilian and Argentine facilities. In March 2002, ABACC inspectors started using SJAR 1.0 in the field with positive results. The field tests and the comments and suggestions provided by inspectors led to further improvements in the software. The English version of SJAR 2.0 was presented in Vienna in November 2002. In December 2002, a period of joint field tests with the IAEA began.

During 2003, ABACC provided training for IAEA and ABACC inspectors and the user's manual of the SJAR 2.0 was compiled in Spanish, Portuguese and English. On September 1st, 2004 the Procedure for Joint Auditing of Records was applied officially in the inspections performed by ABACC and by the IAEA in Argentina and Brazil.

Keywords: inspections, accounting, records, auditing

1. Introduction

In July 1991, Argentina and Brazil signed an Agreement for the exclusively peaceful use of the nuclear energy [1], which established a Common System of Accounting and Control of Nuclear Materials (SCCC) to be applied by the two States. The same Agreement created ABACC to administer and implement this Common System.

In March 1994 a Safeguards Agreement between Argentina, Brazil, IAEA and ABACC [2] entered into force. According to this Agreement, in implementing safeguards IAEA and ABACC shall, to the extent possible, work jointly and coordinate their work to avoid unnecessary duplication of safeguards activities.

Since the beginning of ABACC operations, its accounting unit is responsible for establishing and implementing the procedures for auditing of records, its evaluation and follow-up of discrepancies [3].

A software for use in field inspections was developed, in order to facilitate the completion of working papers by the inspectors avoiding calculations, to allow for the loading of operator's data in electronic media, to obtain results in electronic media, thus facilitating its evaluation and allowing the automate follow-up of discrepancies. The SARA (**S**oftware para **A**uditoria de **R**egistros de **A**BACC) was completed in 1999 and was started to be used on a routine basis in the beginning of 2000. The software has the following advantages:

- organizes the activities to be performed;
- checks all the information entered, for example, MBA codes, algebraic sign of element/isotope weight, etc;
- checks the relationship between different fields to assure that the numeric information entered by the inspector is correct;
- has a mechanism of error detection that does not allow finalizing the auditing if there is a discrepancy between the inspection data and the operator's book data. [4]

The first discussions between ABACC and IAEA concerning their respective methods for records auditing took place in 1997, as a result of the request of the Liaison Committee of the Quadripartite Agreement. In 2000 a decision at the IAEA/ABACC coordination meeting for the implementation of the Quadripartite Agreement led to the two organizations starting to work towards establishing a joint procedure for records auditing.

2. Initial Discussions

In November 2000 the first meeting between the two organizations was held in Vienna. The IAEA explained the procedures for auditing of records applied to facilities in Brazil and Argentina and showed copies of some working papers generated during a few actual inspections. ABACC explained its procedures for auditing of records, presented the SARA software, and showed the working papers generated. The use of actual inspection examples by both organizations was very useful as it permitted the comparison of the results and working papers of each organization.

The results of this comparison showed that both organizations had the same objectives and perform similar activities for records auditing. The main differences in the procedures refers to the ending date of the period to be audited, the format of the working papers and the information generated at the end of the inspection.

It was concluded that for a procedure to be common to both organizations it should be the combination of both present procedures and to attain this objective it was agreed to adopt the IAEA definition for the ending date of the period to be audited and to perform the necessary modification to the SARA software to include the option to generate working papers used by IAEA in hardcopy or electronic media.

With these provisions, the auditing could be performed jointly by both organizations and the results would be obtained in the format suitable for its processing and evaluation by each organization.

Another meeting was held in the beginning of 2001, where new discussions on the procedures of each organization took place, for a better understanding of the information to be generated by the software for IAEA processing and evaluation.

3. Software development

The software SARA was developed in Fox Pro for Windows and allows performing automatically the auditing of records in four main steps:

- step 1: verification that the inventory value at the ending date of the previous audited period remains the same (working paper VR-A),
- step 2: comparison of operator's ledger data with the Inventory Change Reports (ICR) data (working paper VR-B),
- step 3: inputting of the accounting data electronically or manually that were not yet sent through ICRs, including corrections affecting previously audited periods (working paper VR-C),
- step 4: determination of the inventory value at the ending date of the present inspection, based on

data computed during the inspection, and comparison with the ledger value recorded by the operator; for physical inventory verification inspections, the values for book adjusted (BA), physical ending (PE) and material unaccounted for (MUF) are also checked (also VR-A).

To allow a joint use of this procedure, a common ending date of the period to be audited was defined as the last day of the month preceding the inspection, except for physical inventory verification inspections, which is the day of the physical inventory taking.

The mechanism of the procedure was maintained and the necessary modification was introduced in the software SARA to allow the generation of databases for IAEA use and the printing of the working papers used by the IAEA in the auditing of records, namely Accounting Records Examination and Updating (M2, M6) and Comparison of Records with Reports (M5).

In November 2001 ABACC demonstrated the Software for Joint Auditing of Records (SJAR) in Vienna. Inspectors of the IAEA Safeguards Operations B (SGOB) tested the software by performing examples using actual inspection data and presented some comments, in particular with respect to the output generated for IAEA use and processing. Work was also done with IAEA Safeguards Information Technology (SGIT) staff with a view to assess the use of the electronic output of the SJAR to feed IAEA logsheet program, software by which, IAEA inspectors process the information collected in the field.

New modifications were introduced in the SJAR and in March 2002 ABACC inspectors commenced to test the software in the field. The software performed well however the inspectors made a number of comments and suggestions that could improve the software performance. The introduction of these modifications and the inclusion of English as working language in addition to Spanish, led to the version 2.0 of the software.

This version was presented in Vienna in November 2002. In order to evaluate the software operation, the working papers obtained by ABACC inspectors using the software during the testing period, which started in March 2002 were compared with the IAEA logsheets obtained by IAEA inspectors from the same inspections.

The 2.0 version was loaded to IAEA notebooks and a new set of tests was performed using real data of inspections performed in Argentinean and Brazilian facilities. IAEA inspectors presented suggestions to some wording in the software and suggested that a list of tips on SJAR be prepared.

The Agency noted that it would be necessary to compare the electronic ICR data loaded in SJAR (form VR-B) with the ICR data provided by IAEA database, and requested ABACC to provide copies of VR-B before each inspection.

It was noted that the format and the information contained in IAEA logsheets (M2, M6) and (M5) as produced and printed by the SJAR are adequate, except for information concerning the quantity of plutonium for on load reactors shown in form (M2, M6), which should be updated with the fuel discharged from the core and should show the number of items. ABACC should modify the software before the testing period.

Testing was also performed with the IAEA logsheet program and it was concluded that although the electronic output of SJAR were suitable to be input to the logsheet program, this software does not permit the partial loading of modules. IAEA/SGIT would study the problem and try to find a solution.

4. Software Testing

A period of joint field tests with the IAEA started in December 2002, during which the auditing in field was done with the SJAR as well as manually using IAEA logsheets.

In September 2003 training in the SJAR was provided to Agency inspectors in Vienna and a meeting was held to evaluate this joint tests period. It was concluded that the information generated by SJAR for IAEA logsheets (M2, M6) and (M5) was then adequate.

It was noted that there is a need to improve the comparison process of ICR data generated by SJAR with those generated in IAEA database and the Agency suggested to investigate the possibility of electronic comparison. The manual input of updating data (working paper VR-C) was also of concern, especially for facilities with high number of inventory changes and therefore ABACC proposed to agree with the States to provide this information in electronic media, as well as the itemized list of the physical inventory taking.

The testing period was extended and the inspectors continued to provide feedback on the software, which resulted in new improvements. ABACC prepared the draft Procedures for the Joint Auditing of Accounting Records, including instructions for the use of SJAR, and finalized the list of tips for SJAR operation.

5. Joint Procedure Implementation

A new round of training in SJAR and the last evaluation meeting was held in Vienna in April 2004.

For determined facilities having a lot of inventory changes, the operator is presently providing in magnetic media the data of the updating period (VR-C) and the PIT itemized list, which has optimized the time spent in auditing.

It was concluded that it is possible to use the Computerized Inspection On-Site Package (CIOSP) to make the comparison of the electronic ICR data loaded in SJAR (form VR-B) with the ICR data provided by IAEA database. This would be done by loading to CIOSP one of the database files generated by the SJAR. It was noted that it is possible to import IAEA module (M2, M6) into the CIR-Web program by processing one of the files generated by SJAR.

The draft Procedures for the Joint Auditing of Accounting Records was discussed. This document, available in English, Spanish and Portuguese, describes the joint procedures for auditing and has one annex concerning the instructions to operate the software and another where the operational arrangements for the joint auditing are described.

Main provisions determined in the operational arrangements are the following:

- each organization shall provide its inspector with a notebook to perform the auditing; ABACC will be responsible for providing the printer;
- any improvement or modification to the software shall be agreed between the two organizations; the installation and correction of errors shall be done by ABACC;
- inspectors shall be re-trained at least once every two years;
- the list of qualified inspectors shall be exchanged between the two organizations every half a year;
- at least one trained inspector from each organization shall participate in the auditing;
- the auditing is carried out by two inspectors, one of each organization;
- the organization responsible for the operation of the computer in a certain mission is randomly selected at the pre-inspection meeting; the other agency's computer is the spare, to be used in case of any problem with the main computer;
- at the end of the inspection, the inspectors shall go through a menu in the software with allows both organization to have the same information in the hard disk and in the floppy disk.

The official application of the joint procedures started as of September 2004.

In March 2005 ABACC provided another round of training for a new group of SGOB inspectors. At this opportunity the initial training on the SJAR operation was provided to staff of the IAEA Training Section and will be finalized in August 2005. Thereafter, the IAEA Training Section will be responsible for providing re-training to IAEA inspectors, although they are invited to participate in ABACC training to its inspectors in Brazil and Argentina.

6. Conclusion

The joint procedure does not imply an additional effort and brings a significant reduction in the total time dedicated to the auditing, since both organizations are applying a single procedure. The

application of the procedure also led to the national authorities of Brazil and Argentina to provide data in electronic media, which also helped to speed up the auditing activities.

The software SJAR also helped in the detection and correction of mistakes in records and reports increasing the quality of reports provided to the IAEA.

The work developed was possible in such a short time due to the cooperative and positive approach took up by the two organizations, following the guidelines of the Quadripartite Agreement, which advises the joint work between IAEA and ABACC, avoiding duplication of safeguards activities, in accordance with compatible safeguards criteria and reaching its own independent conclusions.

This development is an important step for a future implementation of a full computerized book auditing system from the facility general ledger to the inspection report with all operators providing all data electronically.

7. References

[1] Agreement between the Republic of Argentina and the Federative Republic of Brazil for the Exclusively Peaceful Use of Nuclear Energy. INFCIRC/395. IAEA. Vienna, November 1991.

[2] Agreement between the Republic of Argentina, the Federative Republic of Brazil, The Brazilian-Argentine Agency for Accounting and Control of Nuclear Materials and the International Atomic Energy Agency for the Application of Safeguards. INFCIRC/435. IAEA. Vienna, March 1994.

[3] R.O. Nicolás, “ABACC’s Records Auditing Procedures”, Proceedings of the 19th ESARDA Annual Symposium, Montpellier (1997), EUR 17665, p 723.

[4] R.O. Nicolas, “Software for Records Auditing During Inspections”, Proceedings of the 44th INMM Annual Meeting, Phoenix, Arizona, USA, July 2003.

Session 5

NDA - Advances in Gamma Spectrometry

Limitations of Safeguards NDA Techniques and Challenges for the Future

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Abstract:

Non-destructive assay (NDA) technologies play a crucial role in international Safeguards verification activities. NDA safeguards verification aims to detect quantities of materials, sometimes minimal, often under difficult measurement conditions and to present the resulting data in a clear and straightforward manner for evaluation. This goal is usually achieved by maximizing sensitivity to specific radiation signatures and thus by maintaining high levels of both precision and accuracy.
As a number of external restrictions hamper the free development and implementation of new and enhanced technologies, research and development programs must reconcile the International Atomic Energy Agency needs with these inherent limitations. These limitations arise from nuclear physics principles and other factors encompassing issues such as: portability, overall weight, detection capabilities, user-friendliness, calibration methodology, ease of deployment, accessibility, maintainability, and sustainability.
This paper presents an overview of inherent uncertainties and limitations of some of the current NDA measurement techniques applied for safeguards verification applications. It also reviews the performance expected from these systems, and describes proposed efforts for further improving Safeguards NDA technology.

Keywords NDA methods, NDA techniques, Safeguards

1. Introduction

In performing its safeguards verifications, the International Atomic Energy Agency (IAEA) inspectors rely on NDA measurements. In traditional safeguards, NDA is used to perform an independent check of the Operator Declaration, and in the framework of the additional protocol with complementary access inspections, these methods are used as well to search for undeclared materials, activities and facilities. Current Safeguards NDA measurement technologies rely on the detection of gamma rays, neutrons, heat and ultra-violet light. All methods used can be considered as complementing each other. It is obvious that the sensitivity to radiation signatures is expected to be as high as possible, to obtain the best precision and accuracy, and to allow the optimal detection of minimal quantities of materials, even under difficult measurement conditions.

The free development of such equipment is hampered by several operational limitations that are imposed on the measurement systems, such as portability, weight, counting time (related to detector efficiency), simplicity of calibration, user friendliness, full access to the materials and maintenance. Additionally, basic radiation physics dictates some of the requirements and limitations imposed on the available detection mechanism upon which NDA equipments rely. For example, the lack of discrimination between neutron and gamma radiation due to fundamental limitations in the detection processes of an otherwise attractive detector will restrict its use as Safeguards instrumentation, for which such discrimination has to be good in order to allow a straightforward interpretation of measurement results.

In an attempt to cover this very broad issue, we are presenting some case studies illustrating the limitations faced by some NDA Safeguards measurement techniques and are attempting to provide some insight about potential development path for future NDA Safeguard equipments.

2. Case studies

2.1. Spent fuel measurements

2.1.1 Safeguards Requirements

The IAEA Safeguards Criteria [1] define for each particular stratum of material category, the defect type to be verified, the applicable method and the recommended equipment to be used to reach that goal. In the case of spent fuel stored at nuclear power plants, the verification activities are normally limited to gross defect testing (presence/absence-attribute of material), but in some particular cases (e.g. in the case of inconclusive or conclusive negative C/S, and when required as the material is being moved or has been moved into difficult-to-access storage) a partial defect has to be verified.
Sampling plans for partial defect testing are calculated to detect, with the specified probability, defects totalling 1 Significant Quantity (SQ) in a stratum or group of strata. A partial defect test should therefore be able to detect if a single SQ of nuclear material has been removed from a spent fuel assembly or group of assemblies and possibly replaced by dummies. According to the present safeguards criteria this target amount is half of the fuel pins in a spent fuel assembly. In addition it is important to minimize the probability of false alarms.

2.1.2 Measurement Limitations

An effort was made to investigate if an adequate measurement method exists to detect such possible anomaly. The task was undertaken as a joint effort from 3 Member State Support Programmes to the IAEA and studied the prospects of both a conventional Fork detector (FDET) [2] and an enhanced Fork detector where the gross gamma and neutron signals on a conventional Fork detector is combined with simultaneous gamma spectrometry using a CdZnTe detector.
A FDET configuration was tested in two measurement campaigns by measuring BWR and WWER-440 spent fuel assemblies in interim storage facilities. The investigations showed that a general partial defect test cannot be based solely on independent FDET measurements without making use of the operator's declared data. In the analyses done using only the neutron and gamma measurement data from either conventional or enhanced FDET, there were configurations with pins removed that could not be detected. Using the operator's declared data is necessary due to the influence of both the fuel design and the irradiation history on the measured signals.
It must be mentioned however that the FDET is effective in most real situations that are encountered in inspection practice, although not covering all pin removal scenarios. The main reasons for this are the self-shielding of the gammas and geometry effects that influence the neutron detection. The safeguards limitation in this case comes from the intrinsic limitation imposed by the laws of physics.
To further investigate the partial defect capability of the FDET, measurements were made on a fresh-fuel mock-up (BWR and PWR) to facilitate fuel pin removal. Pins were removed in 3 particular areas of the fuel assembly: central area, by removing rows of rods, and by removing columns of rods, as displayed in Figure 1. The fuel removal scheme is given in table 1. Correction methods were developed to improve the analysis of the results [3].

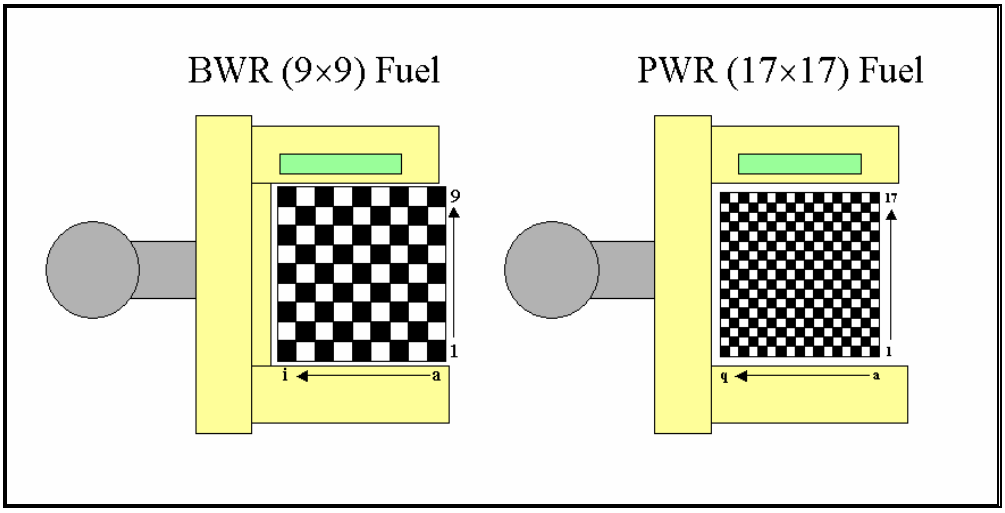

Figure 1: Various regions of pin removal in BWR (9x9) and PWR (17x17) fuel

Table 1: Pin removal pattern in BWR and PWR fuel

Removal pattern	Rows/columns BWR	Rows/columns PWR	Comments
Central part (fig. 1)	8 central rods 24 central rods 48 central rods		Easy to detect
Columns (fig. 1)	9 9,8 9,8,7 9,8,7,6 8,7,6 7,6 6	17 17,16 17,16,15 17,16,15,14 16,15,14 15,14 14	Complex to detect
Rows (fig. 1)	a a-b a-c a-d a-e a-f a-g a-h a-i	a a,c a,c,e a,c,e,g a,c,e,g,i c,e,g,i e,g,i g,i i	Difficult to detect

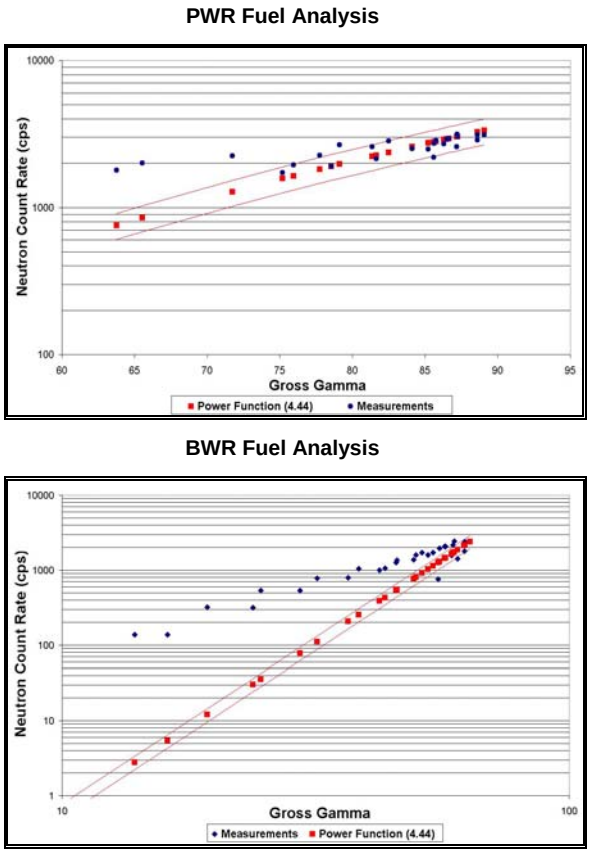

Figure 2: Results (N versus G) for several pin removal scenarios in BWR and PWR fuel

The results of the analysis performed are given in Figure 2 where the neutron count rate is plotted against the gamma signal, and compared with the expression

$$\ln(\dot{\nu}) = a + 4.4 \cdot \mathrm{Ln}(\dot{\gamma})$$

One observes that many configurations where pins are removed the measured data deviate from the general power law, and can be recognized as missing pin items. The regions of pin removal were reported as easy to detect (central), complex to detect (columns 9 to 5), and difficult to detect (rows a to e). This latter case allows making a double measurement with the assembly rotated over 90 degrees. From the data shown in Figure 2, pin removal from a BWR fuel assembly looks easier to detect than from a PWR assembly.

As the results were obtained on fresh MOX fuel, which required for certain hypotheses to be made, some computer simulations have to be performed to confirm the validity of these hypotheses and to clarify the effect of the geometrical configuration of the defect(s). This brought further justification to the general validity of the conclusions.

2.1.2 Future Development

Following advice given during a workshop held at the IAEA in March 2003 [4], and in an attempt to look for complementary or alternative methods, further investigations/development efforts are currently underway. These include R&D activities involving techniques such as tomography, Cerenkov glow observation, gamma and neutron imaging techniques, active neutron interrogation, neutron resonance densitometry, Cd ratio albedo, etc.

Most of these tasks are under partnership with Member States Support Programs, and some of them are already under development.
For example, the Digital Cerenkov Viewing Device (DCVD) [5] was developed as a complementary method to the Cerenkov Viewing Device (ICVD) for fuel with low burn-up (10 MWd/kg) and/or long cooling time (40 years), linked to the detection of weak light sources. The further investigation aims at differentiating between a spent fuel assembly and an irradiated dummy. But in this case, a good visibility in the storage pond is needed, the water quality has to be good, and hindrance from structural materials should be limited, all of them being intrinsic limitations as well.
Each of the proposed methods will definitely have their own intrinsic limitations as well, but their complementarity should help to overcome the practical situations for which there is no real solution, keeping in mind that the real aim is to identify partial defects in spent fuel assemblies.
In all cases, it is most probable that any successful partial defect measurement will require either full or partial isolation of the fuel assembly from the storage rack.

2.2. Portable neutron detector for CA

2.2.1 Instrumentation Concept

The device is intended for use by IAEA inspectors within the framework of inspections carried out for complementary access. With the advent of the HM-5 for detecting the presence of gamma emitting materials, the Agency needs a Portable Neutron Survey Meter (PNSM), which aims at providing a portable neutron sensitive detection solution for this type of inspections.
Performance requirements on the device have been set to provide the highest sensitivity possible to special nuclear material neutrons while answering all other constraints imposed by operability. In terms of transportability, the device should have dimensions consistent with luggage that can be hand carried onto an airplane. Additional limitation on total weight is set to 12 kg. The PNSM should also present very poor sensitive to gamma radiation to provide good discrimination capabilities. The quantity of moderating material available in the device should be adjustable to allow optimisation of the device response for different measurement situations. Batteries need a lifetime of at least 8 hours and portability and transportability should assured in a compact and ergonomic design.

2.2.2 Intrinsic Neutron Detection Capabilities

The Effective Surface Area (ESA) of the device represents a quantity solely dependent of the intrinsic detection capability of the device and it therefore provides a convenient way for evaluating detection performance of neutron detection devices. In the following part of this section we illustrate for a detectors of ESA varying from 20 to 300 cm^2 how the time to reach critical level (identification of a signal as "above Background") varies as a function of distance for a several quantities of fissile material and background count rate. This permits to visualise the intrinsic limitation in detection capability for such type of devices. Calculations were performed by calculating the neutron flux incident onto the detector from the solid angle derived from the relative distance, D, between the neutron source and the detector face. This is shown in Equation 2.

$$\Omega = \frac{1}{4 \cdot \pi \cdot D^2}$$

This quantity multiplied by the ESA and the neutron source strength, $\dot{S}$, yields the neutron count rate for the source, $\dot{\upsilon}$, as shown in Equation 3.

$$\dot{\nu} = \Omega \cdot \dot{S} \cdot ESA$$

Figure 3 shows how the parameters introduced in Equation 2 and Equation 3 can be applied for deriving the intrinsic detection characteristics for a cylindrical detector.

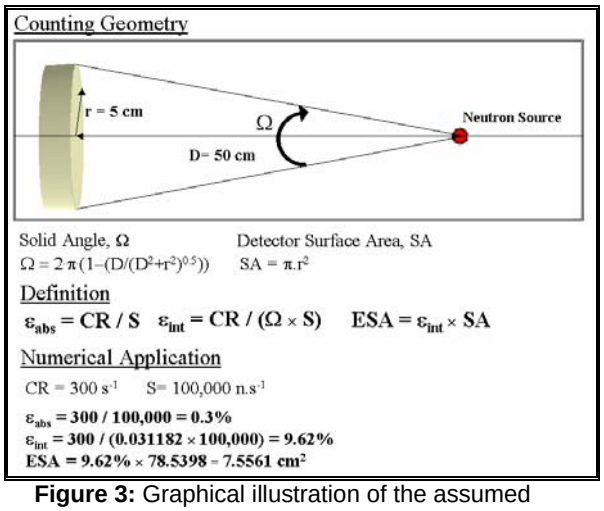

Figure 3: Graphical illustration of the assumed counting geometry

Assuming a given neutron ambient background count rate, $\dot{\nu}_B$, the time to reach critical detection level, T_C, is obtained by checking when the background corrected counts becomes larger that the uncertainty on

the background counts multiplied by the factor function of the chosen confidence level. At 95-% level confidence level this multiplier, k, is equal to 2. Consequently T_C is the minimal value for which Equation 4 is true:

$$T_C \cdot (\dot{\upsilon} - \dot{\upsilon}_B) > k \cdot \sqrt{T_C \cdot \dot{\upsilon}_B}$$

Using this approach allows to quantify the intrinsic performance in terms of detection capabilities of portable neutron detection devices. Figure 5 shows a plot of the variation of time to reach critical level for three neutron counters, respectively with 20, 100 and 300 cm^2 effective surface area for a neutron source strength of 1000 s^{-1} (which is about equivalent to 1-gram ^{240}Pu effective) as a function of distance. The plot shows this evaluation for two ambient background level corresponding to count rates of 0.1 and 0.25 s^{-1}.

2.2.3 Measurement Limitations

The theoretical calculations shown in the previous sub-section illustrate the neutron detection capabilities of devices on term of an intrinsic detection parameter for given measurement situation. It therefore allows the quantification of the detection performance of a device given some design parameters that will limit its intrinsic detection capability. **Error! Reference source not found.** shows, for example, how for a given tube pressure the mass of moderator impacts on the ESA of a device [7]. It indicates that performance gain by adding some moderator reach an optimal compromise. Sensitivity analyses of this type on the different component and design parameter of a device allow obtaining an optimised detection device. It therefore an important part of the design process of new instrumentation.

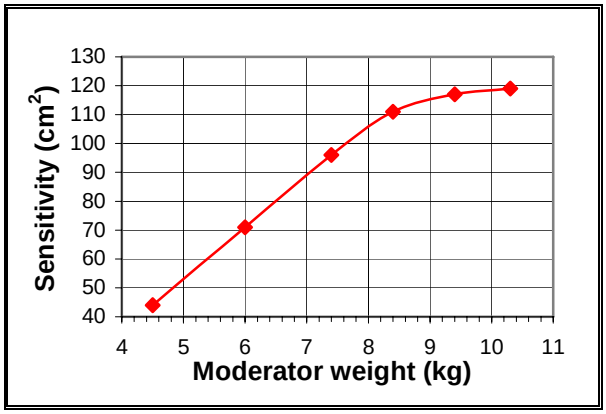

Figure 4: Numerical Sensitivity Analysis of the Detector ESA with mass of moderator for a specific detector design.

Based on the optimisation of a neutron device with operational limitations similar to the ones set for the PNSM it is difficult to imagine a device with an ESA significantly larger than 120 cm^2. Consequently measurement performances for such a device are bound to detection capabilities given by Table 2 for an ambient background of 0.1 s^{-1}.

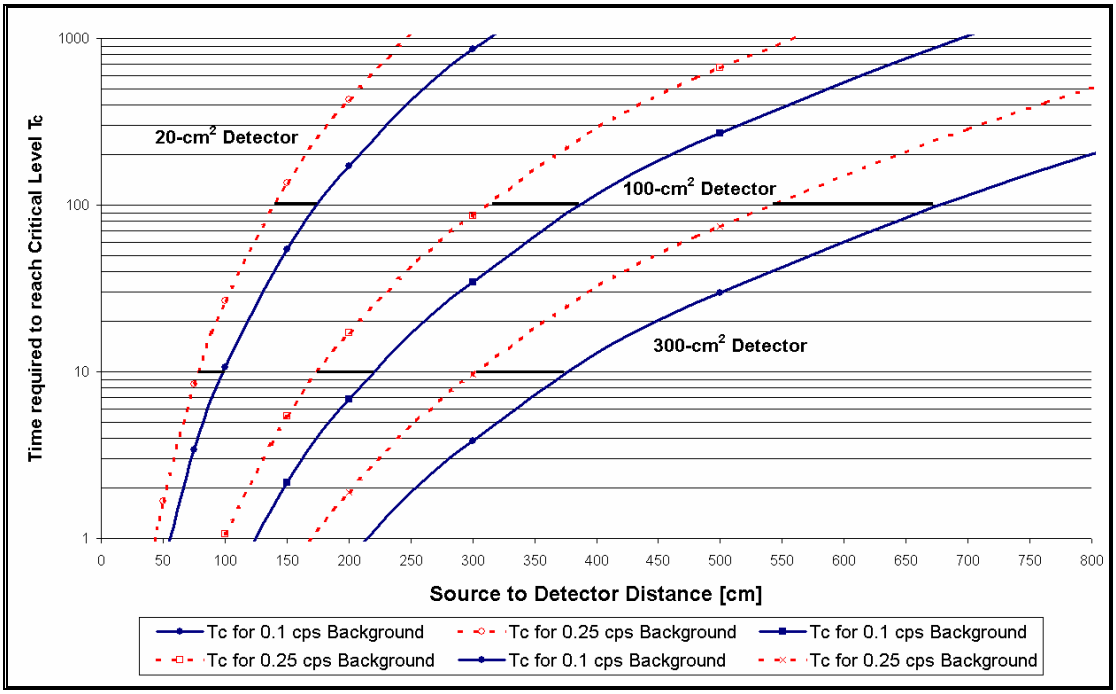

Figure 5: Theoretical estimation of time required to trigger detection as a function of experimental parameters for a 1000 neutron per second source

Table 2: Approximate time in seconds required to trigger a detection alarm as a function of source distance and quantity of unshielded weapons grade (6-% $^{240}Pu_{eff}$) plutonium for an optimised "PNSM" like instrument.

	Source Distance to Detector						
Quantity of Pu	**1-m**	**3-m**	**5-m**	**10-m**	**20-m**	**50-m**	**100-m**
0.001 SQ	2	150	1200	Not Practical	Not Practical	Not Practical	Not Practical
0.003 SQ	<1	17	130	2000	Not Practical	Not Practical	Not Practical
0.005 SQ	<1	6	45	720	Not Practical	Not Practical	Not Practical
0.01 SQ	<1	2	12	180	2900	Not Practical	Not Practical
0.03 SQ	<1	<1	2	20	325	Not Practical	Not Practical
0.05 SQ	<1	<1	<1	8	120	4500	Not Practical
0.1 SQ	<1	<1	<1	<1	30	1100	Not Practical
0.3 SQ	<1	<1	<1	<1	4	125	2000
0.5 SQ	<1	<1	<1	<1	2	45	730
1 SQ	<1	<1	<1	<1	<1	12	180

2.2.4 Potential Improvements

The present study shows how performance of instrumentation based on ^{3}He detectors are intrinsically limited by design restrictions originating from specified operational considerations. However, the performance of a PNSM type instrument as illustrated above constitutes a significant improvement upon the detection capability of small neutron detection hand held devices. The type of portable instrumentation desired is to answer the Agency need to have a capability to detect the presence of an undeclared neutron source with the highest possible sensitivity within a reasonable range and detection time.

2.3. ISOCS (In-Situ Object Counting Software)

2.3.1 Application Description

ISOCS is a deterministic numerical calculation code for the fast determination of the absolute detection efficiency of a characterised high purity germanium detector (HPGe) [9]. Based on a Monte Carlo characterisation of the detector response for gamma ray energy between 45 keV and 7 MeV it allows the evaluation of the absolute response of the detector for a given measurement situation (i.e. geometrical, and chemical description of the measured object and relative position of the detector). The measurement geometry can include user-defined collimators and preset geometry templates covering a wide range of applications. ISOCS has nine standard geometry templates including simple box, complex box, simple cylinder, complex cylinder, pipe, circular plane, rectangular plane, well of Marinelli beaker, sphere.
The method can be applied to a high variety of experimental situations, and in the case of Safeguard verifications, potential applications such as hold-up measurements in bulk material process plants or process waste measurements have been identified.

2.3.2 Measurement Limitations

ISOCS, as any numerical calibration/simulation code, is exposed to several sources of systematic uncertainties affecting its overall accuracy. The Monte Carlo characterisation of the specific detector used for these measurements is a first source of uncertainty. The benchmark of the detector model is performed with a multi-energy gamma point source and distributed sources placed at five generic positions around the detector. Based on that experimental data the Monte Carlo results are quoted to be equivalent to the experimental results to 5-% accuracy with a 68-% confidence level. Although such level of accuracy is acceptable in many waste or

decommissioning activities, it already represents an upper bound for some Safeguards verification activities. Improved method for producing more consistent numerical models could therefore be of use in improving the overall accuracy of the ISOCS calculations.
Combined with this systematic error are all the errors due to the misrepresentation of the true measurement situation made in the geometrical description of the measurements in the ISOCS software. It can be very difficult to practically access the position of the detector relative to the measured object and particular care should be taken at providing deployment tools adapted to the measurement situations.
In order to account for all these uncertainties, the complexity of the summation of their effects makes any arithmetic evaluation nearly impossible (correlated effects). Uncertainties in the dimensions, chemical composition and attenuation properties, density and position of the active material and attenuator materials composing the measured item are not numerically propagated by the ISOCS software so that only user defined uncertainties are associated with the results.
All these remarks mean that ISOCS efficiency results may have very large uncertainties, especially when not well-defined items such as waste or hold-up deposits are measured with no direct possibilities to evaluate the true impact of these input uncertainties on the returned efficiencies. In cases where a high level of uncertainty upon key parameters affecting the measurements is acknowledged arbitrary choices taken when modelling the measured object tend to represent the highest source of inaccuracy.

2.3.3 Potential Improvements

First experience at using the ISOCS for analysing uranium hold-up measurements illustrated some the limitations mentioned above. In order to provide a diagnostic for establishing the validity of the ISOCS modelling assumptions on the thickness, density, shape of uranium deposits in pipes, the possibility to use external information to benchmark ISOCS results was investigated. The use of an isotopic analysis code such as the FRAM software [10] allowed for an independent evaluation of the ^{235}U enrichment to be obtained. Comparing ISOCS enrichment results from the predicted quantities of ^{235}U and ^{238}U present in the hold-up and this external source of information allowed the limited validation of the assumptions taken in ISOCS to model the deposit. Indeed, underestimating the ISOCS ^{235}U enrichment relative to FRAM results corresponds to modelling some cases where the uranium self-attenuation is underestimated (higher relative efficiency at low energy), thus corresponding to interpreting the measured signal as a smaller deposited activity. Consequently, by systematically keeping the ISOCS enrichment below the FRAM measured value the ISOCS analysis conservatively returns a result that can be interpreted as the minimal amount of material present in the hold-up deposit. This approach, which results in potentially large underestimation of hold-up materials could be extended so that more complex diagnostics are performed in order to validate the assumptions taken in the ISOCS modelling. This may involve comparison at various energy regions of the absolute efficiency curve derived with ISOCS with the relative efficiency trend calculated by isotopic analysis codes like MGA of FRAM.
Initial contacts with Canberra Industries may also lead with help from the US support program to an integration of an error propagation scheme in the ISOCS calculation, which would also be a valuable asset to the software.

3. Conclusion

Non-destructive assay measurement techniques are essential tools in safeguards evaluation activities for making independent measurements upon which operator declarations are verified.
Although significant efforts are made to obtain the highest sensitivity, this is sometimes hampered by various kinds of limitations. As external factors, imposed by operational restriction, we consider portability, weight, detector efficiency, user-friendliness, simplicity of methodology, calibration difficulty, facility safety concerns, material accessibility, intrusiveness to operators (e.g. movement of fuel). These external factors are often reflected on other limitations that arise from nuclear physics principles, combined with detector sensitivity, and complexity of geometry.
All these restrictions can hamper free development and implementation of new or enhanced technologies, and research and development programs.
In the present work we tried to illustrate for some applications how by performing either a careful optimisation of instrumentation or by considering the complementarity of other measurement techniques, performance of NDA equipment could be improved. The results of such an analysis is to point out that

in order to meet the challenges set by Safeguards verification criteria, NDA instrumentation and their operational use should be able to reach compromises delivering optimised detection capabilities and complementary measurement techniques allowing for reliable conclusion to be performed in the framework of safeguard inspections.
Only a few examples were treated: spent fuel measurements, a portable neutron detector for rapid unannounced inspections, and the use of ISOCS for the assessment of hold-up in bulk process plants.
The need to improve the NDA equipment used for safeguards verification and homeland security is an ongoing and almost continuous challenge for future development, and underlines the importance of the many collaborations with the Member States.

4. References

[1] IAEA, Safeguards Criteria, SMC.
[2] Rinard PM, Bosler GE, “Safeguarding LWR Spent Fuel with the Fork Detector”, LA-11096-MS, Los Alamos (1988).
[3] Tiitta A, Saarinen J, Tarvainen M, Axell K, Jansson P, Carchon R, Gerits J, Kulikov Y, Lee YG, “Investigation on the possibility to use the Fork Detector for Partial Defect Verification of Spent LWR Fuel Assemblies”, STUK-YTO-TR 191 (September 2002).
[4] IAEA, Coordinated Technical Meeting on Spent Fuel Verification Methods, Vienna 3-6 March 2003.
[5] Chen JD, Gerwing AF, Kosierb R, Axell K, Clarke R, “Cerenkov Characteristics of WWER fuel and non-fuel assemblies using a Digital Cerenkov Viewing Device, INMM annual meeting 2004.
[6] LANL, “MCNP – A General Monte Carlo N Particle Transport Code, Version 5”, Los Alamos National Laboratory, LA-UR-03-1987, April 2003.
[7] Majorov M, private communication (2004).
[8] Currie LA, “Limits for Qualitative Detection and Quantitative Determination, Application to Radiochemistry", Analytical Chemistry, vol 40, 3, (1968),586-593.
[9] "ISOCS Calibration Software“, User’s manual version 4.0, 9231013D, Canberra Industries (2002).
[10] T.E Sampson, T.A. Kelley, "*PC/FRAM version 4.2 Users Manual*", Los Alamos National Laboratory Unclassified Release, LA-UR-99-998, 4th March 2003.

Gamma-spectrometric methods for age-dating of highly enriched Uranium

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Abstract:

Two non-destructive gamma-spectrometric methods for uranium age-dating are presented. The methods rely on measuring the daughter/parent activity ratio $^{214}Bi/^{234}U$ by low-background, high-resolution gamma-spectrometry. The initial methodology was derived during a "Round Robin" exercise, in which the properties of a HEU material relevant to nuclear forensics were assessed by several laboratories. The Uranium-age obtained by this gamma-spectrometric method was in agreement with the results reported by other participating laboratories, which used mass-spectrometry. The original method relied on using an efficiency calibrated geometry, which was easily achieved for thin samples in powder form.
The present work also reports a method for determining the age of Uranium material of any physical form and geometrical shape. This is made possible by using the peaks of ^{238}U for relative efficiency calibration, provided that they can be evaluated from the spectrum. Both methods are shown to be reliable tools for determining the age of uranium samples encountered in nuclear safeguards as well as in illicit trafficking of nuclear materials.

Keywords: age-dating; NDA; low-background gamma spectrometry; HEU

1. Introduction

The age of uranium samples is an important piece of information both for combating illicit trafficking of nuclear materials and for nuclear safeguards. For combating illicit trafficking the age of a seized nuclear material can help in identifying its possible origin. In addition, in view of the Fissile Material Cut-Off Treaty knowing the age of enriched uranium is important for discovering newly produced materials.

The purpose of the present work is to describe two non-destructive, gamma-spectrometric methods for age-dating of homogeneous uranium samples. Until recently, the only method for age dating of Uranium samples was mass spectrometry (see e.g. [1], [2]) which is a destructive method. The gamma-spectrometric methods described in the present paper (see also [3] and [4]) offer simple non-destructive alternatives. The accuracy of the presented methods is close to the accuracy of mass-spectrometric age dating.

The gamma-spectrometric age determination of Uranium samples is based on measuring the activity ratio $^{214}Bi/^{234}U$ by high-resolution gamma-spectrometry in low-background, assuming that the daughter nuclides have been completely removed during last separation or purification of the material [3]. ^{214}Bi is a daughter of ^{234}U, which decays through ^{230}Th to ^{226}Ra, which in turn decays to ^{214}Bi through three short-lived nuclides. The time needed for secular equilibrium between ^{226}Ra and ^{214}Bi to be achieved is about 2 weeks, so it can be assumed that the activities of ^{226}Ra and ^{214}Bi are equal at the time of the measurement. Therefore, using the law of radioactive decay the activity ratio $^{214}Bi/^{234}U$ at time T after purification of the material may be calculated as [3]

$$\frac{A_{Bi214}(T)}{A_{U234}(T)} = \frac{A_{Bi214}(T)}{A_{U234}(0)} = \frac{A_{Ra226}(T)}{A_{U234}(0)} =$$
$$= \lambda_2\lambda_3\left[\frac{\exp(-\lambda_1 T)}{(\lambda_2-\lambda_1)(\lambda_3-\lambda_1)} + \frac{\exp(-\lambda_2 T)}{(\lambda_1-\lambda_2)(\lambda_3-\lambda_2)} + \frac{\exp(-\lambda_3 T)}{(\lambda_1-\lambda_3)(\lambda_2-\lambda_3)}\right] \quad (1)$$

where $A_{U234}(0)$ denotes the activity of ^{234}U at time T=0 while λ_1, λ_2, and λ_3 are the decay constants of ^{234}U, ^{230}Th and ^{226}Ra, respectively. Because λ_1, λ_2, λ_3 <<1, formula (1) can be developed into a Taylor series around T=0, so it can be approximated as

$$\frac{A_{Bi214}(T)}{A_{U234}(T)} = \frac{1}{2}\lambda_2\lambda_3 T^2 . \quad (2)$$

This equation can be used for calculating the age, T, of uranium samples after the activity ratio ^{214}Bi/^{234}U has been determined by gamma spectroscopy.

The activity ratio ^{214}Bi/^{234}U can be determined in several ways from the gamma spectra of the investigated sample. A very precise and reliable way would be to use a reference material of approximately same age, enrichment and form as the investigated sample. The samples encountered in illicit trafficking, however, are usually such that an appropriate reference material cannot be found. Another approach is represented by the method described in [3] that does not require any reference materials but uses the absolute efficiency of the detector determined by “point-like” standard sources. That is, in [3] the activities of ^{214}Bi and ^{234}U in the sample were measured in an efficiency-calibrated geometry. A third approach was followed in [4]: merely a relative efficiency calibration was used to determine the activity ratio ^{214}Bi/^{234}U, without the use of any standard or reference materials.

In the present paper the last two approaches are described on particular measurement examples of determining the age of 90 % enriched and 36 % enriched uranium. The age of the 36%-enriched material was determined by both methods. The results of the different methods coincide with each other within the error limits, confirming the reliability of the methods.

The structure of this paper is as follows. In section 2 the method which uses an efficiency-calibrated geometry is described. The methodologies for determining the activities of ^{214}Bi and ^{234}U are described in subsections 2.1 and 2.2, respectively. The reliability of this method was tested by using a reference material of an approximately known age, as described in subsection 2.3. Section 3 contains the description of the method which uses relative efficiency calibration for uranium age dating. Subsections 3.1 and 3.2 describe how the activity ratios ^{212}Bi/^{238}U and ^{234}U/^{238}U are measured, while in subsection 3.3 the age of the investigated material is calculated. The conclusions are given in section 4.

2. Age dating using efficiency-calibrated measurement geometry

The results presented in [3] were obtained by using 90% enriched uranium oxide powder. In 90% enriched uranium the amounts of ^{234}U and ^{214}Bi are relatively high, so their activity was easily measured. The results of the method described in [3] were confirmed in a Round Robin exercise organized by the International Technical Working Group for Combating Illicit Trafficking of Nuclear Materials [5]. Namely, the value for the age of the investigated material obtained by gamma-spectrometry was in agreement with the values obtained by other laboratories using mass-spectrometry. The method described in [3] is applicable whenever the sample can be approximated to be “point-like” or if it has a well defined geometrical shape, for which an efficiency calibrated geometry can be constructed and for which the self-attenuation can also be accounted for.

2.1. Measuring the activity of ^{214}Bi by low-background gamma spectroscopy

An amount of 0.981 g of the material received within the Round Robin Exercise [5] was placed in a thin, closed polyethylene cylinder of 2.9 cm inner diameter. The gamma-spectrum of the sample was

taken by a coaxial 150 cm^3 high-purity Germanium detector ("PIGC 3520" Intrinsic Coaxial Detector manufactured by PGT), placed in a low-background iron chamber, with wall thickness of 20 cm and 120x60x120 cm (height x width x length) inner dimensions.

First the spectrum of the sample was measured at 6 cm distance from the detector cap for three days. After identifying the 609.3, 1120.3 and 1764.5 keV gamma-lines of ^{214}Bi in the spectrum of the sample, a second measurement was set up in which the sample was placed next to the detector cap for getting a higher count rate. In the second measurement a 4.2 mm thick lead absorber was inserted between the detector and the sample in order to decrease the dead time of the detector caused by the low-energy gamma rays of ^{235}U. The background was monitored in both cases, and it was established that the presence of the lead absorber has no significant influence on the background counts. The count rates of the transitions at 609.3, 1120.3 and 1764.5 keV coming from the sample were about two times of those of the background in the first measurement, and four times in the second measurement.

The activity, *A*, of ^{214}Bi in 1 g of the measured sample was calculated from the count rate per unit mass I (cps/g) by the formula:

$$A = \frac{I}{B\varepsilon} \ , \tag{3}$$

where *B*=0.46 % is the decay branching ratio and ε is the detector efficiency measured by standard reference sources. Because the examined amount of the HEU-containing powder formed a very thin layer on the bottom of the polyethylene cylinder, there was no need for correcting the count rates of the ^{214}Bi lines for self absorption, nor for the possible effect of the movement of the ^{222}Rn-gas within the sample. The activity of ^{214}Bi per unit mass of the sample was found to be 1.9 $\pm$ 0.1 Bq/g as a weighted average of the two measurements.

The agreement of ^{214}Bi activities calculated from the 609.3, 1120.3 and 1764.5 keV gamma-rays proves that these gamma-rays originate, indeed, from ^{214}Bi present in the assayed material. However, because of small branching rations and because detector efficiency decreases with energy, the statistical errors for the counts of the last two gamma rays were significant (50%in the first, and 15% in the second measurement) even for long (3 days) counting times. In addition, these peaks have to be de-convoluted from the peaks of ^{238}U at 1120.6 keV and 1765.5 keV, which also decreases the reliability of the results obtained for the counts at these energies. Therefore, the ^{214}Bi peaks at 1120.3 and 1764.5 keV were merely used for monitoring the origin of the gamma rays, and the 609.3 keV line for calculating the activity of ^{214}Bi which was later used for calculating the age of the sample.

In order to verify that no ^{222}Rn-gas has escaped from the sample while pouring the sample into the container, monitoring the equilibrium between ^{226}Ra and ^{222}Rn was carried out during the measurement by observing the constancy of the count rate of the 609.3 keV line.

2.2. Measuring the activity of ^{234}U

A series of measurements was performed using approximately 0.5, 1, 1.5 and 2 g of the HEU-containing powder. The spectra were taken using a 2000 mm^2 planar HP-Germanium detector (Canberra GL2020R). The detector was standing in a vertical position and the different amounts of the assayed material were placed, one after another, above the detector at a fixed distance of 10.8 cm in the same type of polyethylene container as in the previous measurement. Each of the four spectra was being recorded until the statistical error of the 121 keV line dropped below 1%.

For these measurements there was no need for using the low-background chamber. The values of the count rates of the 121 keV peak of ^{234}U were standardized in unit mass as the ratio, *K(m)*, of the count rate to the total used mass of the sample. Fig. 1 shows the standardized count rates versus total mass. The data were fitted with the function

$$K(m) = K_0 \frac{1-\exp(-am)}{am} \tag{4}$$

which models the law of self absorption within the sample. The parameter K_0 is the value on the vertical axis where the curve crosses the axis and physically it represents the net count rate of 1 g of

the sample corrected for self-absorption. The parameter *a* provides information about the matrix of the sample and its density. A non-linear least-squares fit yields the values $K_0 \approx 4.60$ cps/g and $a \approx 0.65$ g^{-1}.

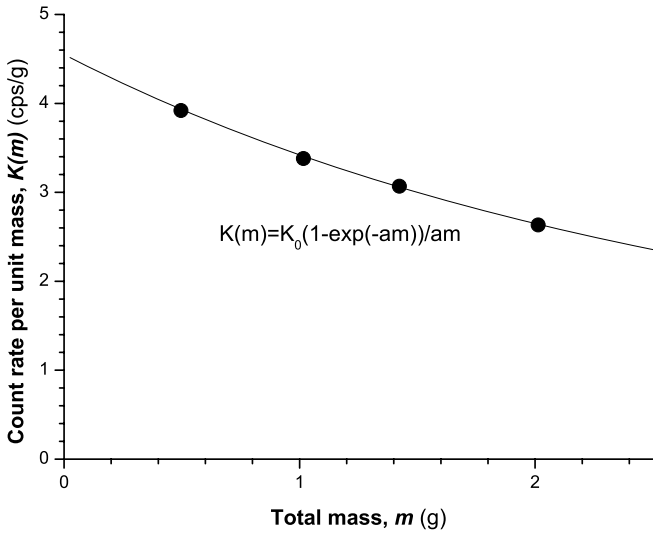

Fig. 1. Count rate per unit mass of the sample, *K(m)*, as a function of the mass of the sample.

The activity of ^{234}U in 1 g of the sample can be calculated from formula (3) inserting $I=K_0$. Using the appropriate decay branching ratio, B=0.0342 %, and detector efficiency, ε=0.730 %, at 121 keV, the ^{234}U activity in 1 g of the sample was obtained as $(1.84 \pm 0.06) \times 10^6$ Bq/g. In this way the activity of ^{234}U was estimated without using any information about the matrix and the isotopic composition of the sample. Note that ^{234}U is a long-lived isotope, with half-life of 2.46×10^5 years, so one may safely assume that its activity at the time of the measurement is the same as its initial activity at the time of production of the assayed material.

Laboratory nickname	Age of the sample (years)
"Azores"	22.2 – 22.6
"Mindanao"	22.4 ± 1.2
"Trinidad"	23.5 ± 0.5
Gamma spectrometric result	23 ± 3

Table 1. The values obtained for the age of the material received in the Round Robin exercise [5], measured by different laboratories using mass spectrometry and the result obtained by gamma-spectrometry

Having obtained the activities of ^{214}Bi and ^{234}U per 1 g of the assayed material, the activity ratio ^{214}Bi/^{234}U was calculated. Then, using equation (2), the age of the material received in the Round Robin exercise was found to be 23 ± 3 years. After the Round Robin Exercise was completed, we received the results reported by other laboratories participating in the exercise, which used mass spectrometry for age dating (see Table 1). The value of 23 ± 3 years is in good agreement with the results obtained by mass spectrometry.

2.3. Verification of the method by determining the age of a reference material

The applicability of the method for Uranium age-dating presented above was tested by applying it for determining the age of a reference material. The reference material was U_3O_8 powder with 90% of ^{235}U. Unfortunately, the exact date of production was not known, but it is documented that the material was transported to Hungary in 1960. The measurement was carried out in 2001, so the material was at least 41 years old at the time of the measurement. Measurements similar to the ones in the case of the Round-Robin material were carried out on this material as well.

An amount of 1.015 g of the reference material was placed into the same type of polyethylene container as the Round-Robin material, and the ^{214}Bi activity was assayed in the described low-background setup. Furthermore, a series of measurements with a varying amount of 0.5 to 2 g of the

material was performed using the planar high-purity Ge-detector described above for estimating the activity of ^{234}U. The spectrum from the low-background measurement is shown on Figure 4, together with the spectrum of the Round-Robin material. Comparing the two spectra one can see that the Compton-background in the spectrum of the Round-Robin material is higher. This is due to the Compton-tail of the gamma-peaks of ^{208}Tl, coming from the decay of ^{232}U which was present in the Round-Robin material. The ^{235}U-enrichment of the two samples was roughly the same, but the reference material was about two times older, so the count rates of the 609.3, 1120.3 and 1764.5 keV peaks of ^{214}Bi were about four times higher for the reference material.

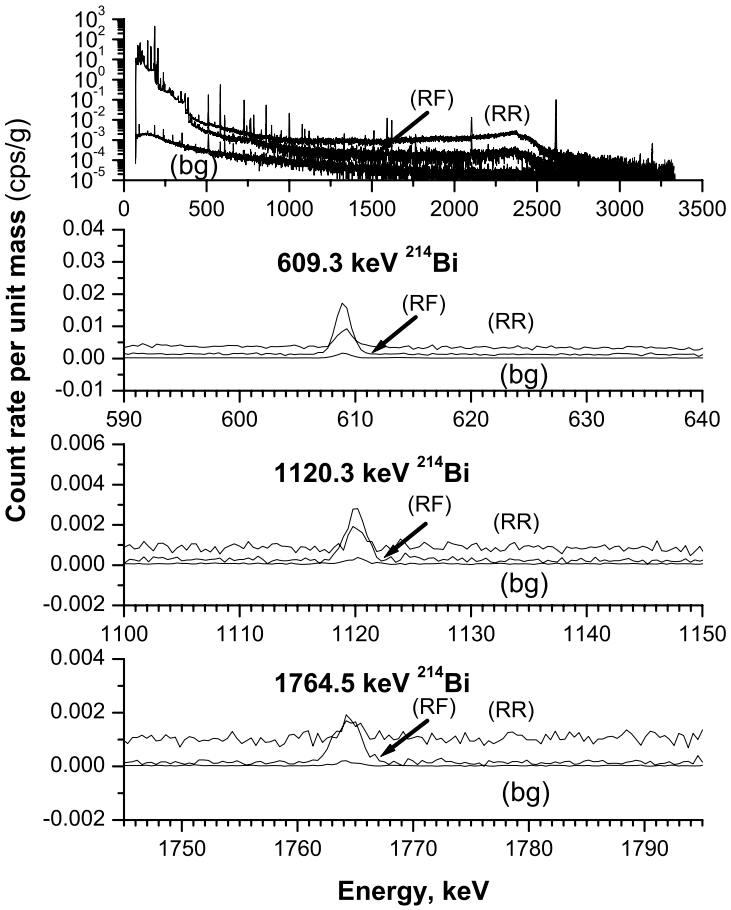

Fig. 2. Gamma-spectrum of the background (bg), the material from the Round-Robin exercise (RR) and the reference material (RF).

The values of the ^{214}Bi activity calculated from the count rates of the three ^{214}Bi peaks agree well with each other, which shows again that these gamma-rays originate from the decay of ^{214}Bi. Because of higher count rates, the relative statistical errors are lower than in the case of the Round-Robin material. Therefore, the activity of ^{214}Bi was calculated as a weighted average of the activities obtained from the three different gamma peaks and it was found to be 7.06 $\pm$ 0.15 Bq/g. The ^{234}U activity was (1.9 $\pm$ 0.06) $\times$ 10^6 Bq/g. The age of the reference material was calculated from formula (2) as 43 $\pm$ 2 years. This value was to be expected from the date of arrival of the material to Hungary. Although no certified reference material of a known age was available, this test gave a good indication of the applicability of the method, even before the results of the Round-Robin Exercise were available.

3. Age dating using relative efficiency calibration

A 36% enriched U_3O_8 powder was used for the measurements. The aim of using a relative efficiency calibration was to develop a method appropriate for age-dating of homogeneous uranium samples of any arbitrary physical form and shape. The idea is to first determine the activity ratios ^{214}Bi/^{238}U and ^{234}U/^{238}U using intrinsic relative efficiency calibration, and then to calculate these results the activity ratio ^{214}Bi/^{234}U. When measuring the activity ratio ^{214}Bi/^{238}U the peaks of ^{234m}Pa (a short-lived

daughter of ^{238}U) are used to construct a relative efficiency curve. The activity of ^{234}U is first determined relative to ^{235}U, and if the enrichment (that is, the activity ratio $^{235}U/^{238}U$ is also known) one can calculate the activity ratio $^{234}U/^{238}U$ as well.

Gamma spectra of the samples were taken in the 0-2 MeV region in low background and in the 0-300 keV region under standard laboratory conditions . From each spectrum a relative efficiency curve was constructed, using the peaks of ^{234m}Pa in the first case, and the peaks of ^{235}U in the second case.

The intensity of the peaks of ^{234m}Pa can be accurately measured within a reasonable measurement time for material of less than approximately 90% enrichment, provided that a sufficient amount of the investigated material is available. It should be noted that for lower enrichments the amount of ^{234}U (and therefore of ^{214}Bi) is lower as well, so the corresponding activity is more difficult to measure and the uncertainty caused by the variation of the natural background becomes more expressed. In addition, a Compton background caused by the peaks of ^{238}U daughters is also present in the spectrum, disturbing the evaluation of the ^{214}Bi peaks. Therefore there exists a lower limit on the enrichment of the material whose age can be determined by gamma-spectrometry, depending, of course, on the amount and the age of the material, detector efficiency and background level.

Four samples of the investigated U_3O_8 powder, having different thicknesses (and therefore different masses), were prepared with masses of approximately 0.5, 2, 5 and 10 g. The four different-mass samples of the material were each held in a thin, closed polyethylene cylinder with an inner diameter of 2.9 cm. The thickness of the layer of U_3O_8 powder on the bottom of the polyethylene cylinder was approximately between 0.04 and 0.92 cm, depending on the mass. For determining the activity ratio $^{234}U/^{238}U$ the spectra of the four samples of the assayed material were taken by planar HPGe detector, whereas the activity ratio $^{214}Bi/^{238}U$ was measured by a coaxial Germanium detector in a low-background chamber.

3.1. Measuring the activity ratio $^{214}Bi/^{238}U$

The activity ratio $^{214}Bi/^{238}U$ was measured in the low-background iron chamber described in the previous section using the same 150 cm^3 coaxial germanium detector. The detector was standing in vertical position and the four different-mass samples were placed, one after another, below the detector so that the bottom of the sample holder was at 6 cm from the detector cap. A 4.2 mm lead absorber was attached to the detector cap, in order to reduce the dead-time caused by the high count rate of the low-energy peaks of ^{235}U. The presence of the lead absorber had no significant influence on the background counts.

The activity ratio $^{214}Bi/^{238}U$ was calculated from the 609.3 keV peak of ^{214}Bi using the relative efficiency curve obtained from the peaks of ^{234m}Pa. ^{234m}Pa is assumed to be in equilibrium with its parent ^{238}U, so it is activity is equal to the activity of ^{238}U. Therefore, the activity ratio $^{214}Bi/^{238}U$ may be calculated as

$$\frac{A_{Bi214}}{A_{U238}} = \frac{I_{609.3} / B_{609.3}}{f(609.3keV)} \tag{5}$$

where A_{Bi214} and A_{U238} denote the activities of ^{214}Bi and ^{238}U, respectively, $I_{609.3}$ is the intensity of the 609.3 keV line of ^{214}Bi, $B_{609.3}$ is its emission probability, while *F(609.3keV)* is the value of the relative efficiency function of the detector at 609.3 keV. The function *F(E)* is obtained by fitting a second order polynomial to the relative efficiencies at the 569.30, 766.37, 1000.99, 1193.69, 1510.20, 1737.73 and 1831.36 keV peaks of ^{234m}Pa.

3.2. Measuring the activity ratio $^{234}U/^{238}U$ and the enrichment

In order to compare different instruments, the low-energy spectra were taken by two planar high-purity Germanium detectors separately (Canberra GL2020 with active diameter of 50.5 mm, thickness of 20 mm and active surface of 2000 m^2 and Ortec GLP-10180/07 with active diameter of 10 mm and length of 7 mm). Both detectors were placed in a vertical position, but the “large” (50.5 mm diameter) detector was looking upward, while the “small” (10 mm diameter) detector was positioned in a downward looking position.

In the case of the "large" detector, the samples were placed, one after another, *above* the detector at a fixed distance of 10.8 cm from the detector cap, while in the case of the "small" detector the samples were placed at 10 cm *below* the detector cap. The gamma peak of ^{234}U at 120.9 keV was observed and the spectrum acquisition lasted until the statistical error of the 120,9 keV line dropped below 2 %. If the sample is not "thin", the activity ratio ^{234}U/^{235}U of the material can be reliably determined using the "U235" [6] or the "MGAU" [7] code. For thin samples, such as the samples used in this research, however, the results of the U235 and the MGAU codes contain large systematic errors (see e.g. [8]). Therefore, in this research these codes were not used for determining the ^{234}U content of the material.

Using relative efficiency calibration the activity ratio ^{234}U/^{235}U was determined, as

$$\frac{A_{U234}}{A_{U235}} = \frac{I_{120.9} / B_{120.9}}{f(120.9keV)} \tag{6}$$

where A_{U234} and A_{U235} denote the activities of ^{234}U and ^{235}U, respectively, $I_{120.9}$ is the intensity of the 120.9 keV line of ^{234}U, $B_{120.9}$ is its emission probability, while *f(120.9keV)* is the value of the relative efficiency function of the detector at 120.9 keV. The function *f(E)* is obtained by fitting a second order polynomial to the relative efficiencies at the 143.8, 163.3, 185.7 and 205.3 keV peaks of ^{235}U.

The enrichment of the material was determined from the spectra taken by the small planar detector, using the commercial implementation of the "U235" code, which is part of ORTEC's software package "MGA++" [10]. The values obtained for the enrichment of the sample did not depend on the thickness of the sample, confirming the findings of [8]. Therefore the enrichment was calculated as the weighted average of the values obtained by measuring the enrichment of the four different-mass samples.

When the enrichment is known (and therefore also the activity ratio ^{235}U/^{238}U) the activity ratio ^{234}U/^{238}U can be simply calculated as

$$\frac{A_{U234}}{A_{U238}} = \left(\frac{A_{U235}}{A_{U238}}\right) \times \left(\frac{A_{U234}}{A_{U235}}\right). \tag{7}$$

3.3. Calculating the age of the material from the measured activity ratios

In order to calculate the age of the sample, one needs the activity ratio ^{214}Bi/^{234}U. It is calculated from the results of the previously described measurements as

$$\frac{A_{Bi214}}{A_{U234}} = \frac{(A_{Bi214} / A_{U238})}{(A_{U234} / A_{U238})}. \tag{8}$$

The age of the material was calculated from equation (2) for each sample separately, as presented in Table 2. The values $T_{1/2}(^{230}\text{Th})=7.538\times10^{4}$ years and $T_{1/2}(^{226}\text{Ra})=1600$ years taken from reference [10] for the half life of ^{230}Th and ^{226}Ra, respectively, were used in the calculations.

Mass (g)	^{214}Bi/^{238}U (Bq/Bq)	Activity ratios ^{234}U/^{238}U (Bq/Bq)		Age (years)	
		Small det.	Large det.	Small det.	Large det.
0.52	$2(1)\times10^{-4}$	106(4)	101(3)	32(10)	33(10)
2.33	$3.7(6)\times10^{-4}$	100(2)	99(9)	43(4)	43(6)
5.55	$3.6(5)\times10^{-4}$	97(6)	99(10)	43(5)	43(5)
10.49	$3.6(3)\times10^{-4}$	97(7)	102(9)	43(4)	42(4)

Table 2. Results of the measurements obtained using four different-mass samples of the investigated material.

It can be seen from Table 2 that the results obtained using different amounts of the investigated material agree very well with each other, except for the case of the 0.52 g sample. In the case of the smallest sample the error of determining the count rate of ^{214}Bi is very high, because in this case the relative contribution of the background to the total count rate at 609.3 keV is very high (about 70%). That is, the error of the count rate at 609.3 keV is dominated by the fluctuations of the background

(about ±15%). Accordingly, the results of repeated measurements with the smallest sample placed at 6 cm distance from the detector have shown a big dispersion. The influence of the background can be lowered by placing the 0.52 g sample closer to the detector, so that the relative contribution of the background to the total counts becomes smaller. Therefore, another measurement with the 0.52 g sample was made, but this time placing the sample closer to the detector, in such a way that the sample holder was touching the led shielding. In this measurement the relative contribution of the background to the total counts of the 609.3 keV line was about 50 % and the activity ratio $^{214}Bi/^{238}U$ was calculated to be $4 \pm 1 \times 10^{-4}$ Bq/Bq, resulting in 46 ± 8 years for the age of the sample. Although this value is close to the value obtained by using larger amounts of the investigated material, its uncertainty is still quite high. Therefore, the measurement of the age of small-amount samples and a detailed study of background fluctuations deserve further attention.

For comparison, the age of the investigated material was also calculated by using the absolute efficiency of the detector, as described in section 2. In order to account for the self absorption of the 120.9 keV line within the sample, the procedure described in subsection 2.2 was applied. In the case of the 10 g sample the count rate is the highest, so in this case the relative uncertainty produced by the fluctuation of the background level is the lowest. Therefore, only the results of the measurements with the 10 g sample were used for calculating the activity of ^{214}Bi in this comparison. The age obtained using the absolute efficiency of the detector is 45 ± 4 years, which, within the error limits, coincides with the value obtained using the relative efficiency calibration.

4. Conclusion

In this paper two methods for the gamma-spectrometric age-dating of uranium samples are described. Both methods use the activity ratio $^{214}Bi/^{234}U$ as a “chronometer” for calculating the age of the material. These methods do not require the use of reference materials of known ages. For the first method it is necessary to know the absolute efficiency of the detector, so the measurements can only be performed in an efficiency-calibrated geometry. In contrast, the second method, using merely a relative efficiency calibration, is appropriate for age dating of uranium samples in arbitrary measurement geometry. The accuracy of these methods is close to the accuracy of mass-spectrometric age dating. The “difficult” samples are the same as in the case of mass spectrometry: low-enriched and/or “young” uranium. The sensitivity and accuracy of the presented methods depends on the background level, detector efficiency, measurement time and the enrichment and amount of the assayed sample. Using the coaxial HPGe detector and the low-background iron chamber described above, the lower bound for age determination of 90 % enriched HEU was estimated to be around 5 years. The sensitivity and the range of applicability of the methods may be improved by using a detector with a higher efficiency, by further lowering the background level and by minimizing the background fluctuations. If a well-type HPGe detector could be used, the sensitivity of the method could be improved, so that the lower bound for the applicability of the method would be approximately 1 year for 90% enriched Uranium and 20 years for natural Uranium, under the present background conditions.

It can be concluded that gamma-spectrometric age dating of uranium is, within the described limits, a reliable tool for determining the age of uranium samples encountered in combating illicit trafficking of nuclear materials and in nuclear safeguards.

Acknowledgement

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References

[1] Wallenius, M., Morgenstern, A., Apostolidis, C., Mayer, K., *Determination of the age of highly enriched uranium*, *Anal. Bioanal. Chem*. **374**, (2002), 379-384

[2] Morgenstern, A., Apostolidis, C., Mayer, K., *Age Determination of Highly Enriched Uranium: Separation and Analysis of ^{231}Pa*, *Anal. Chem*. **72** (2002) 5513-5516
[3] Tam Nguyen C., *Age-dating of highly enriched Uranium by gamma-spectrometry*, *Nucl. Instrum. Meth. B* **229** (2005) 103-110
[4] Tam Nguyen, C., Zsigrai, J., in preparation (to be submitted to *Nucl. Instrum. Meth. B*)
[5] Dudder, G. B., Hanlen, R. C., Herbillon, G. H., *HEU RR Status*, Presentation given at the Seventh Meeting of the International Technical Working Group for Combating Illicit Trafficking of Nuclear Materials, June 27-28, 2002, Luxembourg
[6] Clark, D., *U235: A Gamma Ray Analysis Code for Uranium Isotopic Determination*, UCRL-ID-125727, (1996)
[7] Gunnink R. *et al*, *MGAU: A New Analysis Code for Measuring U-235 Enrichments in Arbitrary Samples*, UCRL-JC-114713, (1994)
[8] Berlizov, A. N., Tryshyn, V. V., *Study of the MGAU Applicability to Accurate Isotopic Characterization of Uranium Samples*, IAEA-SM-367/14/05/P
[9] MGA++ Software User's Manual, Software Version 1, EG\&G ORTEC
[10] Ekstrom, L. P., Firestone, R. B, *WWW Table of Radioactive Isotopes*, database version 2/28/99, http://ie.lbl.gov/toi/ (2004)
[11] Kocherov, N., Lammer, M., Schwerer, O*., Handbook of Nuclear Data for Safeguards*, INDC(NDS)-376, IAEA, (1997)

Design of COMPUCEA 2nd generation for simplified in-field uranium inventory verification measurements

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Abstract:

The equipment for COMPUCEA, a measurement technique for combined uranium element and ^{235}U enrichment assay, has been re-designed with the goal of simplifying in-field applications. Specifically, radioactive sources and detectors requiring cooling with liquid nitrogen have been eliminated from the existing COMPUCEA equipment without sacrifice of measurement performance. The revised measurement approach and instrumentation is described in this paper.

Keywords: uranium concentration measurement, ^{235}U enrichment measurement, L-edge densitometry, high-resolution gamma spectrometry, low-resolution gamma spectrometry.

1. Introduction

Analysts from the Institute for Transuranium Elements (ITU), Karlsruhe, provide continuous in-field measurement support to Euratom and IAEA safeguards during the physical inventory verification (PIV) in different European low-enriched uranium (LEU) fuel fabrication plants. For this purpose they are executing on-site analytical measurements with COMPUCEA (COMbined Product Uranium Concentration and Enrichment Assay), a transportable gamma spectrometry device capable of measuring both the uranium element concentration and the ^{235}U enrichment in samples from uranium product materials. In order to achieve both types of measurements, COMPUCEA incorporates a K-edge densitometer for the uranium element assay, and a passive high-resolution spectrometry measurement of the 186 keV gamma ray from ^{235}U in a well-controlled measurement geometry for the determination of the ^{235}U enrichment /1,2,3/.

Originally COMPUCEA had been designed for the analysis of uranium solution samples. For this type of sample COMPUCEA is allowing a straightforward nondestructive assay. In practice, however, the majority of the samples to be analysed during PIV campaigns in LEU fuel fabrication plants are of solid form (powders, pellets), for which the COMPUCEA measurement requires a prior sample dissolution. Because of this additional step of sample preparation, an assay with COMPUCEA represents more a kind of radioanalytical analysis involving careful analytical procedures (like quantitative dissolution, solution density measurements, quantitative aliquoting etc.) than a mere spectrometry measurement. The respective in-field analyses with COMPUCEA during PIV campaigns have been so far carried out by well-trained analysts from ITU. From this measurements it could be demonstrated that the accuracy of the present COMPUCEA measurements is well within the performance limits set by the International Target Values for this kind of spectrometry measurements /4/, i.e. 0.2% for the element assay and 0.5% for the ^{235}U enrichment determination in LEU materials.

The design of a 2nd generation of COMPUCEA equipment was therefore not primarily aiming at an improvement of the current measurement performance. Instead, the re-design of the instrumentation has been mainly driven by some practical considerations. The current version of the COMPUCEA equipment uses a relatively strong radioactive gamma source (a mixed ^{57}Co-^{153}Gd source with about $5 \cdot 10^8$ Bq) for the K-edge densitometry measurement, and two HPGe detectors requiring on-site cooling with liquid nitrogen. Because of the presence of the radioactive source, and of the size of the current equipment, a special instrument transport has to be arranged so far for each on-site mission.

Further, because of the required detector cooling, one extra mission day is normally foreseen for on-site instrument preparation.

In order to somewhat alleviate the kind of logistical problems encountered in the in-field measurements with the current COMPUCEA equipment, and with the expected benefits of saving some time and costs for the missions in mind, ITU is presently developing a 2nd generation of the COMPUCEA device. The underlying development work also constitutes a task in the support programme of the Joint Research Centre of the European Commission to the IAEA. This paper provides an interim report on the achievements made, with the focus set on the assay in COMPUCEA dealing with the uranium element analysis.

2. Design criteria

The development of a 2nd generation of instrumentation for COMPUCEA has been started with the following main design goals in mind:

- Eliminate the radioactive source(s) to simplify transportation;
- Down-size the weight and volume of the equipment for true portability;
- Avoid detector cooling with liquid nitrogen to save inspection time and operator support;
- Retain current analysis procedure of performing the uranium element assay and ^{235}U enrichment measurement on the same sample solution to minimise sample preparation;
- Conserve the existing measurement performance as far as possible.

To meet the above criteria, the following main changes have been taken into consideration:

- Substitution of the present K-edge densitometry measurement for the uranium element assay by a corresponding L-edge densitometry measurement;
- Replacement of the HPGe well detector currently used for the ^{235}U enrichment determination by a NaI scintillation well counter with improved software for data analysis.

3. Uranium element assay

3.1 Choice of method and equipment

Gamma or X-ray absorptiometry (with the unavoidable step of sample dissolution for solid samples) will remain the preferred non-chemical analysis technique for uranium element assay as long as true nondestructive methods for high-accuracy element analysis in small samples are still lacking in NDA measurement technology. If radioisotopes are to be excluded as radiation source, resort has to be taken to X-ray generators as a convenient type of photon source. For portable equipment this excludes K-edge densitometry as measurement option because of the bulk of the 160 kV X-ray generators needed for this purpose.

By contrast, only 25-30 kV X-ray generators are needed for L-edge densitometry on uranium with his L_{III}-absorption edge occurring at 17.17 keV. Small and compact X-ray generators of this type are now commercially available. L-absorption edge spectrometry offers in addition the advantage that for the spectrometry of the relatively low-energy photons Peltier-cooled Si detectors can be preferably used as X-ray detectors instead of liquid nitrogen-cooled Ge detectors. This has led to a relatively compact measurement configuration and equipment for the uranium element assay in the 2nd generation of COMPUCEA as shown in Fig. 1. The main equipment components include:

- A compact X-ray generator with maximum ratings of 30 kV and 100 μA (model Eclipse-II from Amptek Inc.) with a Ag target transmission tube and associated controller;
- A Peltier-cooled 10 mm^2 x 0.5 mm Si Drift Detector system with associated preamplifier/amplifier and power supply (model AXAS-SSD10 from KETEK GmbH) for high pulse rates, offering an energy resolution of 142 eV at 5.9 keV at a pulse shaping time constant of 0.5 μs Gaussian equivalent;
- A digital signal processor (model DSA 1000 from Canberra);
- A shielding/collimation assembly with sample adapters allowing both L-edge densitometry and XRF measurements.

The total weight of this uranium element assay part of the new COMPUCEA, including the shielding for the X-ray tube, amounts to about 6 kg (without Laptop), which meets the design goal of true portability.

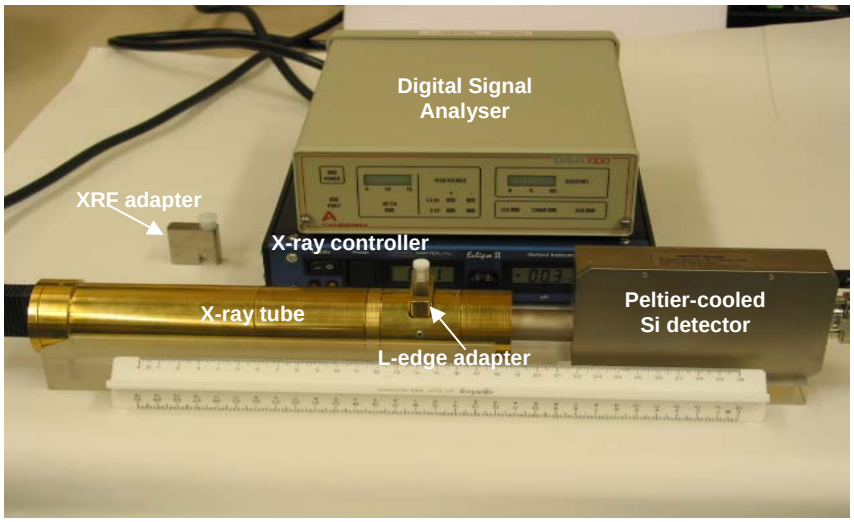

Fig. 1. Photograph of the L-edge/XRF densitometer for the new COMPUCEA.

3.2 Some design details

The irradiation assembly has been designed for two types of measurements: (i) for L-edge densitometry as main operating mode, and (ii) for optional XRF measurements. The XRF option has been added to provide measurement capabilities for the control and quantitative determination of relevant matrix constituents such a gadolinium, which is usually present in some of the uranium samples to be analysed at notable concentrations (up to 10 wt%). A fairly good knowledge of this particular additive appears desirable in order to allow for eventual bias corrections to be made on the L-edge densitometry assay result for uranium.

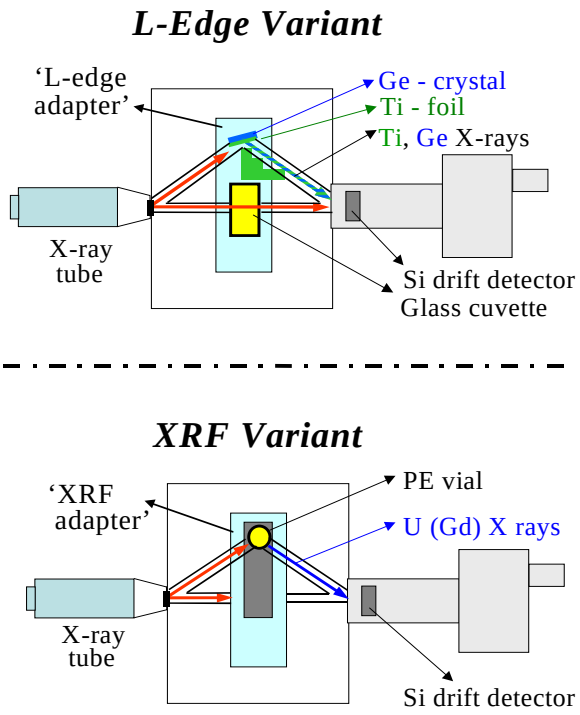

Fig. 2. Irradiation assembly for alternative L-edge densitometry and XRF measurements.

The adopted configuration for L- edge densitometry is schematically shown in the top diagram in Fig. 2. A collimated straight-through beam from the X-ray tube is passing through a removable precision quartz cuvette containing the uranium measurement solution towards the Si Drift Detector. Some compromise had to made as to the path length of the sample cuvette according to one of the above design criteria, calling for concentration and enrichment measurements to be made on the same sample solution for reasons of practical simplicity. For the enrichment assay in COMPUCEA the amount of uranium in the measurement sample should be as high as practically possible because of the low specific gamma activity of ^{235}U. In the present COMPUCEA measurements the uranium concentration in the measurement samples is adjusted to about 200 mgU/ml for the parallel K-edge densitometry and ^{235}U enrichment measurements, which is optimal for K-edge densitometry and, at the same time, still provides reasonable counting rates for the enrichment measurement. To maintain the same concentration level for L-edge densitometry would imply the use of a measurement cell with a path length of not more than 2 mm because of the much higher photon attenuation above the L_{III} absorption edge of uranium (see Fig. 3). However, the use of (disposable or re-usable) 2 mm cuvettes has been ruled out because of their associated relative uncertainty for the path length (0.5% relative for the most accurate cuvettes available), which would directly propagate as uncertainty into the concentration assay result.

For the time being we have adopted as a compromise removable 5 mm cuvettes as measurement cell, for which the uncertainty in the cell depth reduces to 0.25% relative (Note: To further reduce this uncertainty component, the cell depth of a given set of cuvettes could be easily determined relative to each other with an accuracy of 0.1% better from suitable L-or K-edge densitometry measurements, for example, from K-edge measurements performed on a zirconium solution of appropriate concentration). With the use of a 5 mm cuvette the practically useful uranium concentration level has to be reduced to about 100 gU/l or slightly less (see Fig. 3). Fig. 4 shows a typical measurement example for the L-edge spectrum obtained from a 74 gU/l uranium solution contained in a 5 mm cell. The X-ray tube is operated at 24 kV. The uranium concentration is determined from the jump at the L_{III} edge, which offers the largest differential change in the photon transmission.

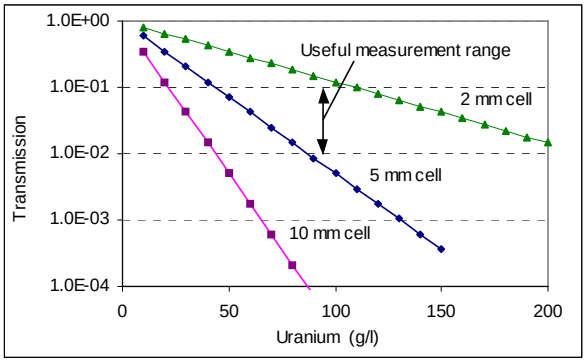

Fig. 3. Photon transmission above the L_{III} edge as a function of uranium concentration and solution depth.

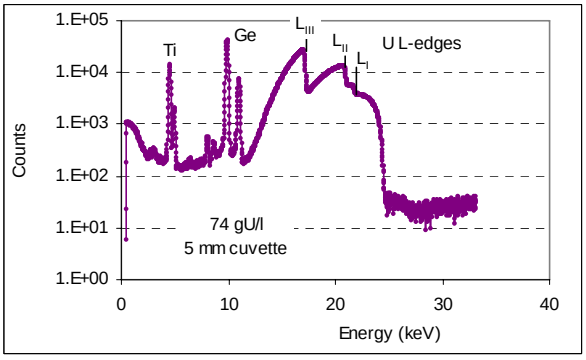

Fig. 4. L-absorption edge spectrum from a uranium solution with added X rays from Ti and Ge.

The L-edge spectrum in Fig. 4 also shows the presence of characteristic K X rays from Ti (Ti-Kα,β = 4.508/4.931 keV) and Ge (Ge-Kα,β = 9.874/10.978 keV), which have been added to the spectrum for the purpose of regular energy calibration and detector resolution monitoring. The reference lines are generated from a fluorescence assembly consisting of a poly-crystalline Ge platelet covered with a thin Ti metal foil, which is irradiated from a second X- ray beam from the X-ray tube as shown in Fig. 2. The added reference lines have been placed into a spectral region where they do not disturb or interfere with the uranium L-edge spectrum.

A permanent control sample in the form of a 0.05 mm thick zirconium metal foil mounted in a Ti holder for insertion into the L-edge adapter has been prepared for the purpose of QA measurements. The K-

absorption edge spectrum from zirconium (E_K = 17.998 keV) provides a comparable measurement situation as the L-edge absorption spectrum from uranium as illustrated in Fig. 5.

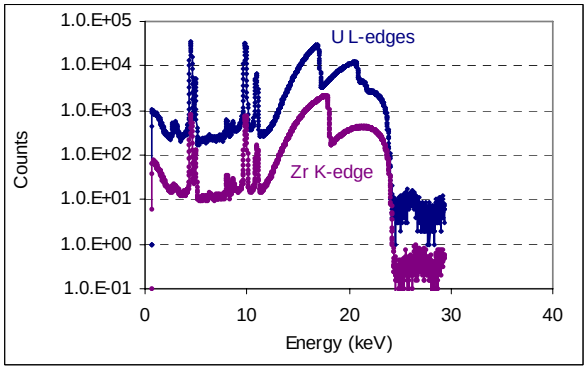

Fig. 5. Absorption spectra from a uranium solution and a zirconium metal foil.

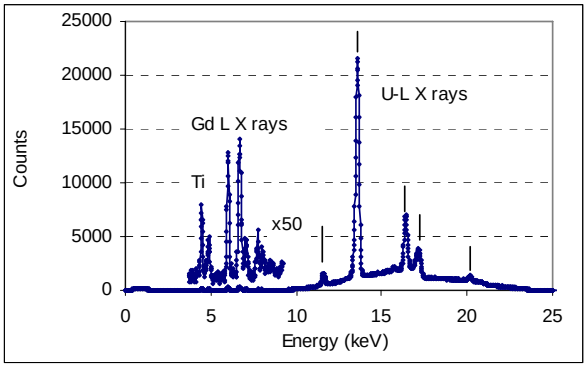

Fig. 6. XRF spectrum from a uranium solution with 10 wt.% of gadolinium.

In order to switch from the L-edge densitometry mode into the XRF analysis mode, the 'L-edge adapter' accommodating the 5 mmm quartz cuvette and the Ti/Ge fluorescence target has to be exchanged for an 'XRF adapter', which can accommodate a thin-walled cylindrical PE vial as sample container for the XRF measurement (see Figs 1 and 2). The XRF adapter is directing an X-ray beam from the tube towards the PE vial but is blocking the straight-through beam. Fig. 6 shows a measurement example obtained from a uranium solution containing 10 wt% of Gd. The XRF analysis mode is only used for measuring the element ratio of matrix elements relative to uranium. The determination of the Gd/U ratio is determined from the measured intensity ratio of the Gd $K_{\beta 1}$ X-ray at 6.712 keV and the U L_l line at 11.622 keV, which are closest in energy.

3.3 Data evaluation

A new COMPUCEA software has been written to manage the data acquisition, data evaluation, calibration, and measurement assurance for the L-edge densitometry measurements. The evaluation of the uranium concentration from the measured ratio of photon transmission across the L_{III} edge at 17.17 keV follows the proven analysis procedure adopted for K-edge densitometry with an X-ray continuum /5/. In this approach the photon transmission as a function of energy, T(E), is measured relative to a blank spectrum from a nitric acid solution of representative molarity, and then linearised in a representation lnln(1/T) *vs* lnE. Linear least-squares fits to the respective data on both sides of the absorption edge determine the photon transmission at energies slightly displaced from the absorption edge ('non-extrapolated fitting mode'), or directly at the absorption edge energy ('extrapolated fitting mode'). Fitting intervals ranging from 15.5-16.7 keV, and from 17.6-18.8 keV were chosen for the evaluation of the transmission ratio across the L_{III} edge.

4. Test measurements

A number of parameter studies and test measurements have been carried out to study relevant measurement features such as the impact of matrix effects, suitable measurement range and linearity of the measurement response, measurement precision and repeatability, performance of key instrument components etc. Some pertinent results are briefly discussed below.

4.1 Matrix effects

Two kinds of matrix effects of practical relevance have been studied: (i) the influence of the acidity level of the HNO_3 solvent, and (ii) the impact of a Gd additive on the measured uranium concentration. The dependence on both parameters has been determined experimentally and compared with the existing measurement situation in COMPUCEA using the 103/122 keV gamma pair form a mixed ^{153}Gd/^{57}Co source for the uranium concentration measurement.

Fig. 7 shows that the determination of the uranium concentration with the new L-edge densitometer becomes -as expected- virtually insensitive to matrix effects, if the transmission ratio is determined directly at the L-edge energy from the extrapolated fitting mode. In the non-extrapolated fitting mode

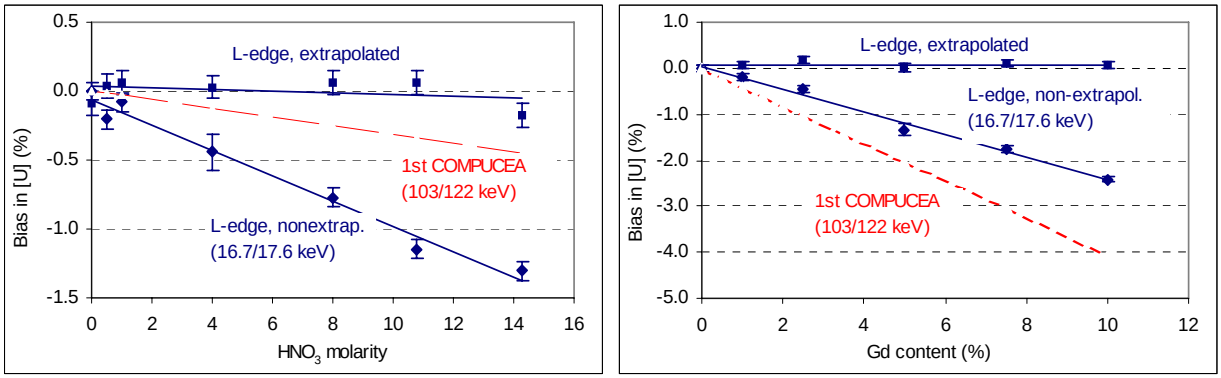

Fig. 7. Percentage bias effect due to variations in HNO_3 acidity (left) and Gd content (right). Assumed uranium reference concentrations: [U] =100 g/L for L-edge densitometry and [U] = 200 g/L for 1st COMPUCEA.

determining the photon transmission at energies before and after the edge (16.7 and 17.6 keV, respectively) the bias effect increases for the molarity case but decreases for the Gd case in comparison to the situation in the present COMPUCEA. In any case, the adopted procedure in the new COMPUCEA of evaluating the uranium concentration from both the non-extrapolated and extrapolated fitting mode provides a very powerful tool for spotting and quantifying eventual bias effects caused by the solution matrix. Further support for the quantification of higher Z matrix elements can be obtained from the optional XRF analysis mode.

4.2 Measurement range and precision

Measurements performed on a set of reference solutions (25-145 gU/l) proved that the useful upper concentration limit (at a cell depth of 5 mm) should not significantly exceed the level of 100 g/l. Above this level the photon transmission above the uranium L_{III} edge decreases to values of less than 10^{-2} (see Fig. 3), with the consequence of nonlinearity effects starting to develop Fig. 8). Uranium concentrations between about 50-100 g/l represent an optimum measurement range both in terms of response linearity and measurement precision. The measurement precision plotted in Fig. 9 for the non-extrapolated and extrapolated fitting mode refers to the presently adopted beam collimation with a collimator diameter of 1mm, which provides total counting rates in the L-edge spectrum of about 6000-10000 cps.

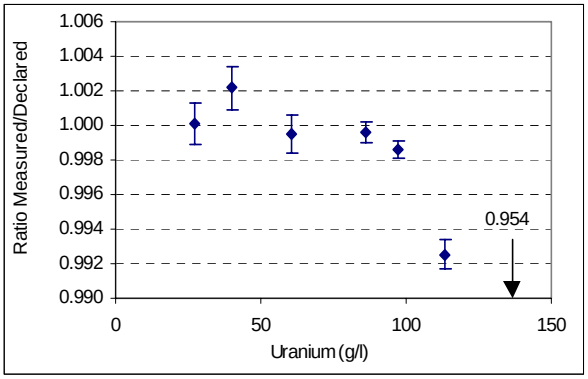

Fig. 8. Calibration of the L-edge densitometer with uranium reference solutions.

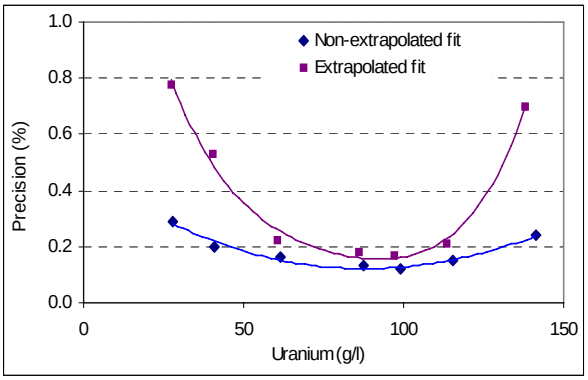

Fig. 9. Measurement precision for the uranium concentration (counting time 2000 s)

4.3 Stability of the X-ray system

A main prerequisite for high-accuracy absorption edge densitometry measurements with a continuous X-ray beam is a high degree of voltage stability of the high voltage supply for the X-ray tube. Since pertinent technical specifications were not available for the miniaturised (and relatively low cost) X-ray generator selected for the new COMPUCEA equipment, the test and qualification measurements performed with the new equipment also had to address this issue.

The actual high voltage supplied to the X-ray tube is easily determined with high accuracy from the endpoint energy of the X-ray continuum in the L-edge spectrum. This determination is part of the routine analysis program for measurement control and assurance. Respective measurement data from a 2-months period (Fig. 10) show a high voltage drift of not more than 0.025 kV (0.1% relative), which fully meets the requirements. On the other hand it was also observed that the photon output from the X-ray tube, expressed as counting rate in the transmitted L-edge beam, varied in the same period by more than 20% (Fig. 11). Fortunately, this variation in photon output is of no relevance both for the L-edge densitometry and XRF measurements in the new COMPUCEA, because both types of measurements are simply based on intensity ratio measurements.

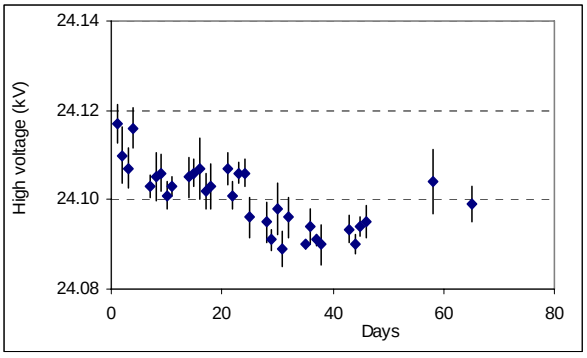

Fig. 10. High voltage stabilty of the X-ray generator.

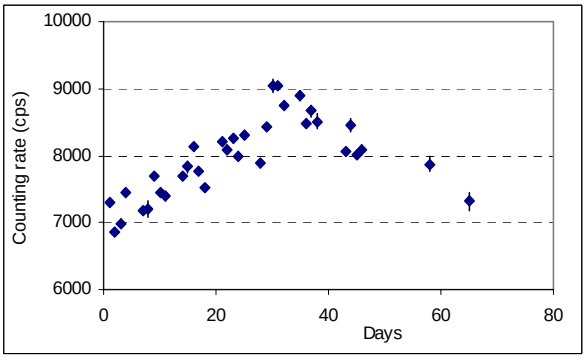

Fig. 11. Variation in counting rate as a result of fluctuations in the photon output from the .

4.4 Measurement repeatability

An integral equipment test for the assessment of the overall instrument stability and measurement repeatability has been carried through extended repeat measurements on a uranium solution sealed in a 5 mm quartz cuvette, and on the Zr metal foil sample prepared for QA measurements. The results from 36 (37) repeat measurements performed over a period of 70 days are plotted in Fig. 12. Each data point represents the mean of 5 consecutive repeat measurements of 2000 s each, providing on the average a statistical counting precision of 0.080% for the L-edge densitometry measurement on the uranium solution, and of 0.072% for the K-edge densitometry measurement on the Zr foil. The standard deviations for both measurement series are fully compatible with the figures for the statistical counting error, which proves the absence of other notable instrument variabilities beyond counting statistics during the test period.

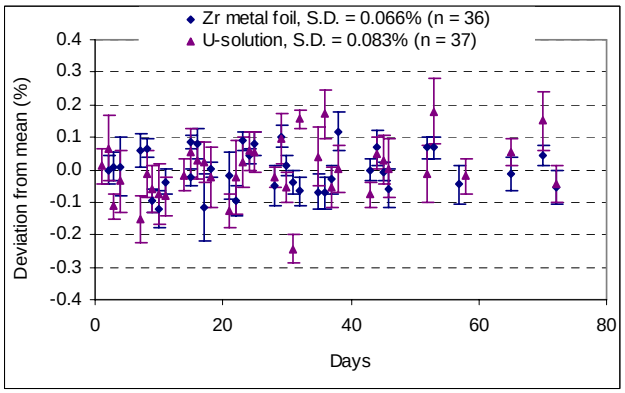

Fig. 12. Measurement repeatability observed during a period of 70 days for measurements on a sealed uranium solution (87 gU/l), and a Zr metal foil.

5. ^{235}U enrichment assay

The ^{235}U enrichment measurement in COMPUCEA is based on the counting of the 186 keV gammas from ^{235}U from a defined amount of uranium in solution form in a defined counting geometry as illustrated in Fig. 13. The standard detector used so far is a 110 cm^3 HPGe well detector with a well diameter of 16 mm in the detector cap. The measurement procedure requires accurate aliquoting of 2.5 ml of the sample solution into a cylindrical plastic vial for counting in the well detector. The accurate amount of uranium in the measurement sample is calculated from the known uranium concentration of the solution sample determined in the parallel K-edge densitometry measurement, from the solution density obtained from an additional density measurement, and from the accurately controlled volume aliquot of the sample. At the typical uranium concentration of 200 mgU/ml this means an amount of approximately 0.5 g of uranium used for the enrichment measurement, which provides a source strength for the 186 keV gammas from ^{235}U of roughly 200-250 photons per sec and percent enrichment. For the given counting configuration with a measured photo peak detection efficiency of 27 % at 186 keV for a 2.5 ml solution sample this results in a counting rate of 50-60 counts per sec and percent enrichment for the 186 keV line.

The calibration for this kind of enrichment determination in COMPUCEA is made with well-characterised uranium reference solutions. The calibration also takes into account a small non-linearity of the measurement response as a function of the uranium concentration. With the detection rate of 186 keV photons mentioned above the statistical counting error normally represents the dominating uncertainty component for the typically adopted counting times of 1000-2000 s.

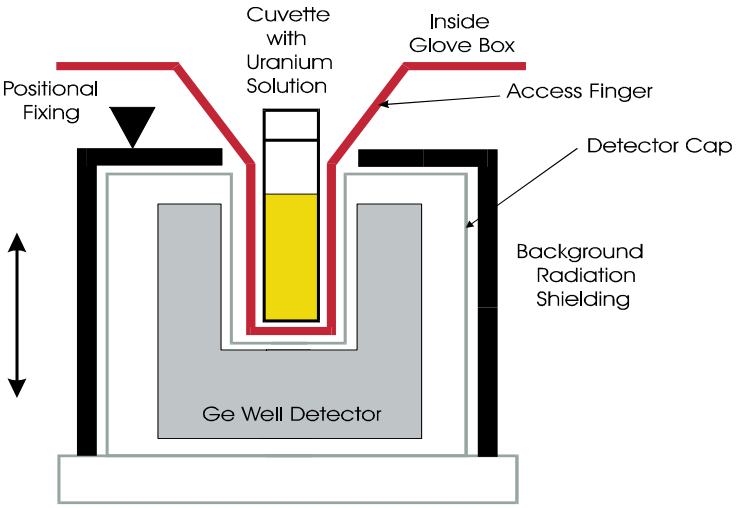

Fig. 13. Counting configuration for the ^{235}U enrichment measurement in COMPUCEA (Note: For in-field measurements the glovebox enclosure indicated in the figure is replaced by a thin-walled cylindrical sleeve of PE inserted into the detector well).

The proven procedure for the enrichment measurement will be also retained in the 2nd generation of COMPUCEA. The only modification considered is to replace the Ge detector by a detector operating at room temperature, i.e. to use a NaI scintillation well detector of similar or identical well dimensions instead of the HPGe well detector. We have tested a 7.6 x 7.6 cm NaI well detector with a well diameter of 16 mm and a measured resolution of 6.5 % at 662 keV and 9.0% at 186 keV. A measurement example taken from a 4.3% enriched uranium sample is shown in Fig. 14. The given NaI detector provides a more than two times higher detection efficiency (58% for the 2.5 ml sample) compared to the Ge well detector, but suffers of course from the much poorer energy resolution. For an accurate determination of the net 186 keV peak area it is proposed to use the method of response function fitting such as provided, for example, by the NaIGEM (NaI Gamma Enrichment Measurements) code, which has proved to provide bias-free net 186 keV peak areas over the full range of enrichment from natural to highly enriched /6/. An example for a spectrum fit with the NaIGEM code is shown in Fig. 15. The method of response function fitting is also capable of coping with specific properties of NaI-photomultiplier based detector systems, namely gain shifts and changes

in detector resolution caused by variations in ambient (temperature) conditions. We are quite confident that under the well-controlled measurement conditions prevailing in the COMPUCEA application the use of the NaI detector system in combination with the advanced spectrum analysis will turn out to represent a viable alternative to the measurements with a Ge detector. Pertinent laboratory and in-field tests are in preparation.

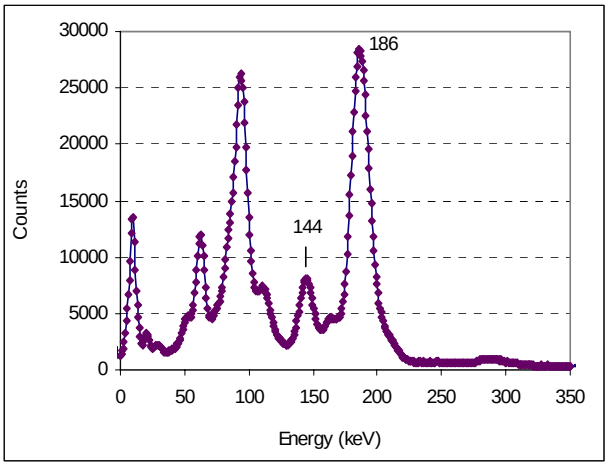

Fig. 14. Gamma spectrum measured with a 7.6 x 7.6 cm NaI well detector from a 4.3 % enriched uranium sample.

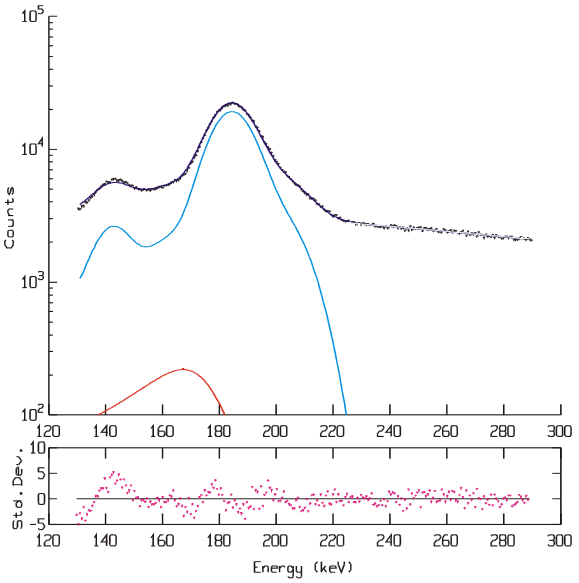

Fig. 15. Example of a response fit with the NaIGEM analysis code (by courtesy of R. Gunnink).

6. Conclusions

The principal design criteria for a 2nd generation of COMPUCEA equipment, i. e. the elimination of radioactive sources and nitrogen-cooled detectors, have been fully achieved and demonstrated for the uranium element assay in COMPUCEA. The result is a compact and truly portable instrument, which will certainly help to simplify to some extent the logistical aspects for in-field measurement campaigns. It is important to note that the simplification in terms of measurement equipment has been achieved without any compromise in terms of measurement performance. The true performance evaluation will be subject of the forthcoming in-field campaigns, where it is planned to operate the 2nd generation of COMPUCEA in parallel with the existing version.

For the ^{235}U enrichment assay part in COMPUCEA it is proposed to replace the high-resolution Ge detector by a NaI scintillation detector as one of the main measure for simplifying the equipment. With the use of improved software for spectrum analysis it is expected to be able to greatly maintain the measurement performance presently achieved with the high-resolution Ge detector. This expectation still remains to be verified.

7. Dedication

This paper is dedicated to the memory of Peter Matussek, who deceased on 24 February 2005. The development of the COMPUCEA measurement technique and equipment in its early stages, and its successful demonstration in various in-field exercises, is intimately related to his name. It has been mainly due to his scientific dedication and his continuing demands for high standards in measurement technology that COMPUCEA has gradually evolved to one of the most accurate radiometric analysis techniques in Safeguards. Likewise, his thorough and exhaustive treatment of the gamma-spectrometric ^{235}U enrichment measurement technique, laid down in the User's Manual for the Certified Reference Material EC-NRM-171/NBS-SRM-969 (KfK 3752, 1985) and completing the first important action of the ESARDA NDA Working Group, remains a lasting legacy of Peter Matussek for the gamma spectrometry community.

8. References

[1] P. Matussek, I. Michel-Piper, H. Ottmar, “A Compact Densitometer for Uranium”, Proc. 13th ESARDA Symp., Avignon, France, 14-16 May, 1991, 329.

[2] P. Matussek, I. Michel-Piper, H. G. Wagner, B. Brandalise, O. Cromboom, “First In-Field Experience with COMPUCEA – A Portable Assay System for the Combined Determination of the Uranium Concentration and ^{235}U Abundance in Uranium Solutions by NDA Methods”, Proc. 15th ESARDA Symp., Rome, Italy, 11-13 May, 1993, 415.

[3] P. Matussek, Ph. Laurent, W. Janssens, F. MacLean, U. Blohm-Hieber, “Routine Use and Recent Developments of the COMPUCEA Instrument”, Proc. 19th ESARDA Symp., Montpellier, France, 13-15 May, 1997, 313.

[4] International Target Values 2000 for Measurement Uncertainties in Safeguarding Nuclear Materials, Report STR-327, International Atomic Energy Agency, Vienna, April 2001.

[5] H. Ottmar, H. Eberle, “The Hybrid K-Edge/K-XRF Densitometer: Principles-Design-Performance”, Report KfK 4590, Kernforschungszentrum Karlsruhe, February 1991.

[6] R. Gunnink, R. Arlt, R. Berndt, “New Ge and NaI Analysis Methods for Measuring ^{235}U Enrichments”, Proc. 15th ESARDA Symp., Rome, Italy, 11-13 May, 1993, 431.

NaI Gamma Spectrometry of Shielded Uranium / Limits and New Approach of Uranium Identification

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Abstract

The detection of uranium is of greatest interest because it is the type of nuclear material so far mostly encountered in illicit trafficking. However, its detection and identification is generally difficult and sometimes even impossible, especially if the uranium is highly enriched and concealed under shielding. In this paper we will present a complete study of the uranium gamma spectra seen by a typical hand-held monitor with a NaI detector through various shielding materials. Resulting gamma spectra were both obtained experimentally and by Monte Carlo simulation studies. From a review of the spectral data we propose a new approach for the identification of uranium which makes also use of the shape of the measured gamma continuum observed from shielded uranium. This complementary information can be helpful in cases of poor statistical data for the full photon energy peaks of uranium gamma rays. The applicability and limitations of this analysis approach will be discussed.

Keywords:
Illicit trafficking, Uranium, shielding, gamma spectrometry.

1. Introduction

On 17-11-2003 at 22h08 we received the following e-mail from a colleague in xxxx, *" I need your help. Border police stopped truck from xxxx, in the post you visited previously. I went with hand held device (detector NaI) to identify source. The Hand held device could not identify and wrote "Not in library". Trailer was very big. I measured around it. Dose rate in one place was 2-3.5 microSv/h, but in all the rest was background dose rate level. I send you the spectrum. I think it can be bremsstrahlung from beta source, is it? Please have a look." Best regards"*. The material was in fact shielded uranium, which could not be detected with the mentioned instrument.

Nowadays fighting against illicit trafficking of nuclear materials is as important as never before. At the customs only hand-held devices can be used. NaI and CZT detectors are the most commonly applied hand-held spectrometers, as they are compact, easy-to-use and need minimal support requirements. On the other hand their resolution capability is limited. The detection of uranium is of greatest interest because it is the type of nuclear material so far mostly encountered in illicit trafficking.

Gamma ray spectra from uranium samples depend on the enrichment. The 185.7 keV gamma ray is the most prominent single gamma ray from a uranium sample enriched above natural U-235 level. In the case of natural and low enriched uranium the 766.4 keV and 1001 keV gamma lines from Pa, a daughter of U-238, are also useful for identification. The identification and the determination of enrichment can be carried out easily when no other isotopes and shieldings are present.

The identification of shielded radioactive or nuclear sources by gamma spectrometry is still an unsolved problem. Hand-held devices used by custom officers at borders suffer from this limitation. The shape of the spectra of a shielded source is specific to the nature of the radioactive isotopes of the source and the nature and thickness of the shielding materials. Highly enriched uranium is the most difficult nuclear material to be identified when it is shielded. ^{235}U is emitting gamma lines at low energies (up to approximately 200 keV). Daughter products from the ^{238}U are emitting gamma lines at high energies (766.4 and 1001 keV). It is then more easy to detect depleted than enriched shielded uranium. However, an important contribution to the U spectra

comes from beta particles from the decay of ^{234m}Pa emitted with maximum energy 2.29 MeV. The beta particles are producing bremsstrahlung in the uranium source itself. The Bremsstrahlung continuum is representing a significant contribution to the shape of the measured spectra.

In this paper we will present a complete study of the uranium gamma spectra seen by a typical hand-held monitor with a NaI detector through various shielding materials. Resulting gamma spectra were both obtained experimentally and by Monte Carlo simulation studies. From a review of the spectral data we propose a new approach for the identification of uranium that makes also use of the shape of the measured gamma continuum observed from shielded uranium. This complementary information can be helpful in cases of poor statistical data for the full photon energy peaks of uranium gamma rays. The applicability and limitations of this analysis approach will be discussed.

2. Methods

2.1 Simulation software

The Monte Carlo code MCNP ver.4 was used for simulations of the transport of photons in a 3D space in a geometry consisting of various cells with different materials. The following interactions are taken into account for the transport of photons: photoelectric absorption with possibility of emission of X-ray fluorescence photons, coherent (Thompson) scattering modified by scattering functions, incoherent (Compton) scattering modified by form factors and pair production. The cross sections of these interactions are contained in libraries for elements from hydrogen (Z=1) to plutonium (Z=94). An electron library is available for the electron transport. Several output tallies – current, flux across a surface, track length estimate of the flux in a cell are available. One of the features is a pulse height tally, so it can simulate a spectral energy distribution of particles detected by a detector.

2.2 Detector

We assumed that the detector is a NaI(Tl) detector with a 1x1" cylindrical crystal (2.54 cm diameter and height as well) as typically implemented in hand-held devices. The detector is covered by a MgO reflector and enclosed by aluminium on the one side, and on the other side there is a glass coupling to a photomultiplier. The layer of MgO is 1 mm thick, and the Al housing of the detector is 0.8 mm thick.

2.3 Geometry of the problem

The following three spherical geometries were chosen for the source material:

- 2 cm diameter, density 10 g . cm^{-3}, mass 41.88 g of UO_2, (pellet)
- 5 cm diameter, density 5 g . cm^{-3}, mass 327.25 g of UO_2, (pellets)
- 9 cm diameter, density 2.5 g . cm^{-3}, mass 1000 g of UO_2, powder.

The source is positioned in a centre and is surrounded by air. The assumed lead shielding is built as a layer of variable thickness at 5 cm from the centre, and later as a box with walls at 5 cm from the centre. The lead thickness considered was 1 and 5 mm. First simulations were done with one NaI(Tl) detector with its front side at 10 cm distance from the centre. The simulations were improved by introduction of six identical detectors positioned on main axes of the geometry (x, y, z).

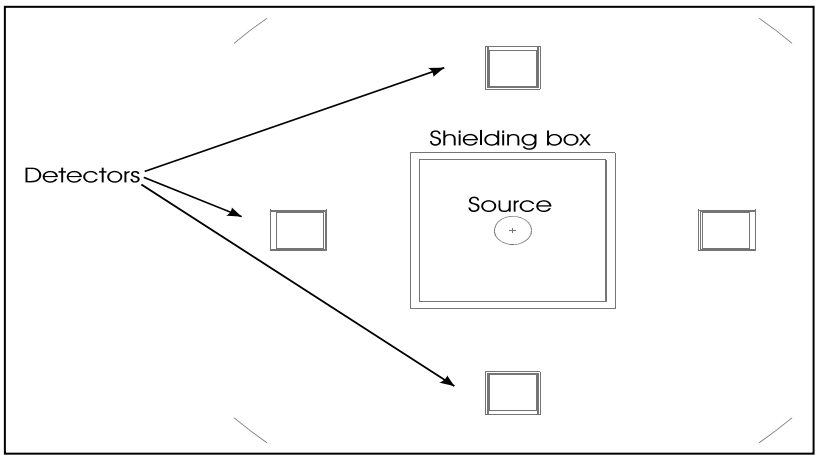

Fig. 1. The x-y cut of the geometry with six detectors. Spherical UO_2 source is in the middle. Two remaining detectors are on the z axis. The thickness and material of the shielding is determined by the thickness of the shielding box and can change.

2.4 Photon energies

Photons produced by the radioactive decay of uranium are largely shielded by the uranium material itself, with a high degree of absorption mainly at low energies below 300 keV. The number of photons flying out of a real sample is then different from the total number of photons produced in a source. The analysis of the importance of photons of different energies for detection therefore has to take into account the self-absorption in the source. It is obvious that the photons of higher energies are less absorbed.

The self-absorption was calculated using the f2 tally – flux averaged over a surface for energies in an interval 30 – 2000 keV. The self-absorption in the spherical source with diameter 2 cm and the density 10 g . cm^{-3} versus the photon energy is shown in Fig. 2.

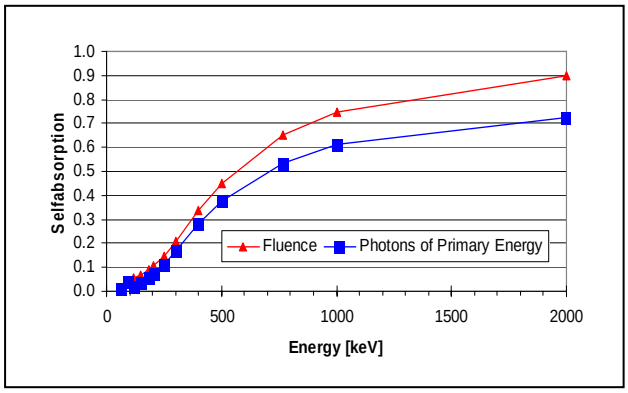

Fig. 2. Self-absorption of photons in the source of 2 cm diameter with a UO_2 density of 10 g·cm^{-3}. The points were calculated using MCNP and represent the number of photons emitted from the sphere per one starting photon of corresponding energy. Triangles are representing the primary and scattered photons, and squares are the photons of primary energy only.

The uranium isotopes and their daughter products emit a variety of gammas and X-rays. The importance of the different radiations depends on their emission probability and on the self-absorption in the source as well. The relevance of the different gamma lines for detection is changing when we take into account the enrichment and self-absorption. The number of escaping photons per gram then determines to:

$$Y = A_g \cdot EP \cdot SA,$$

where A_g is the specific activity [Bq · g^{-1}] of uranium nuclides (for daughter products equilibrium with parent U isotopes is assumed), EP is the emission probability and SA is the self-absorption. Table 1 lists the number of escaping photons from a UO_2 sphere (ϕ 2 cm, density 10 g·cm^{-3}), ranked by intensity, for 4 different enrichment grades.

Table 1. Number of primary photons (Y) emitted from a UO_2 sphere with diameter of 2 cm and density of 10 g · cm^{-3} for four different enrichments in ^{235}U.

	HEU 90%			LEU 3.50%			Natural 0.72%			Depleted 0.2%		
No.	E [keV]	Nuclide	Y	E [keV]	Nuclide	Y	E [keV]	Nuclide	Y	E [keV]	Nuclide	Y
1	185.7	U-235	2389	185.7	U-235	93	1001.0	^{234m}Pa	63	1001.0	^{234m}Pa	66
2	205.3	U-235	263	1001.0	^{234m}Pa	61	766.4	^{234m}Pa	19	766.4	^{234m}Pa	20
3	143.8	U-235	240	766.4	^{234m}Pa	19	185.7	U-235	19	92.8	Th-234	14
4	93.4	U-235	173	92.8	Th234	14	92.8	Th234	14	92.38	Th-234	14
5	84.2	Th-231	156	92.4	Th234	14	92.4	Th234	14	63.29	Th-234	7
6	163.3	U-235	156	205.3	U-235	10	63.3	Th234	6	185.7	U-235	6
7	105.0	U-235	99	143.8	U-235	9	742.8	^{234m}Pa	5	742.8	^{234m}Pa	5
8	90.0	U-235	98	93.4	U-235	7	786.3	^{234m}Pa	3	786.3	^{234m}Pa	4
9	109.2	U-235	61	84.2	Th231	6	205.3	U-235	2	112.8	Th-234	2
10	202.1	U-235	55	163.3	U-235	6	112.8	Th234	2	1737.7	^{234m}Pa	2
Bremsstrahlung			37			390			401			403

Gammas in the low energy region are the most important ones for high enriched uranium (HEU). With increasing amount of ^{238}U in the materials, high energy gammas from ^{234m}Pa at 1001 and 766.4 keV are becoming more and more important. For the assessment of the number of detected photons, however, also the dependence of the detection efficiency of the gamma detector on the gamma energy has to be taken into account. The final shape of the spectrum as measured by the detector is depending also on the amount of scattered photons in the source itself, in materials surrounding the measuring geometry, and in materials present between the active detector volume and the source. The most important items are the materials covering the detector (aluminium and magnesium oxide), and materials of an eventual shielding of the source.

2.5 Bremsstrahlung

Three isotopes are decaying by emission of beta particles with a decay probability of ~100% .Two of them (^{234}Th and ^{231}Th) are producing beta particles of relatively low energy with a maximum at 0.2 and 0.3 MeV, the third one (^{234m}Pa) has a maximum of the energy distribution at 2.29 MeV. These beta particles are producing bremsstrahlung inside the material of a source. The mean of the energy distribution of the bremsstrahlung continuum is approximately at 1/3 of the mean of the beta particles energy distribution. The low energy electrons from the $^{231,234}Th$ decay are producing bremsstrahlung mainly at low energies, with the mean of the distribution far below the 100 keV. In contrast, the mean energy of the betas coming from ^{234m}Pa is 0.82 MeV. The bremsstrahlung photons are then produced with the mean of ~ 250 - 300 keV. These photons are partly absorbed by the material of the source, so the final mean energy is shifted to higher energies (approximately 400 keV). The actual mean energy slightly depends on the shape and the density of the source.

The production of the bremsstrahlung in a UO_2 source was obtained by the simulation of the electron transport in the source. The continuous beta distribution of beta decay (Fig. 3) needed as input was modeled as a histogram (20 steps in energy), and electrons were accounted to be emitted with isotropic distribution inside the volume of the source. The bremsstrahlung photons flying out of the UO_2 sphere were then tallied in 10 keV energy bins. (Fig. 4).

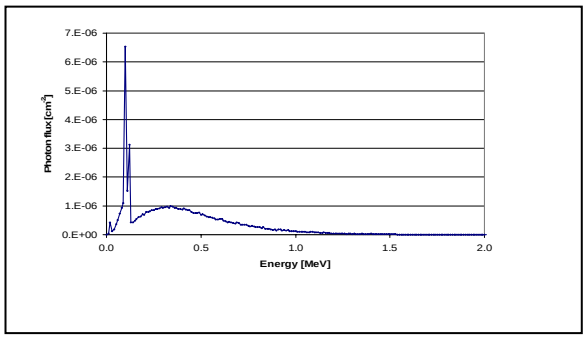

Fig. 3. Spectrum of beta particles emitted by ^{234m}Pa.

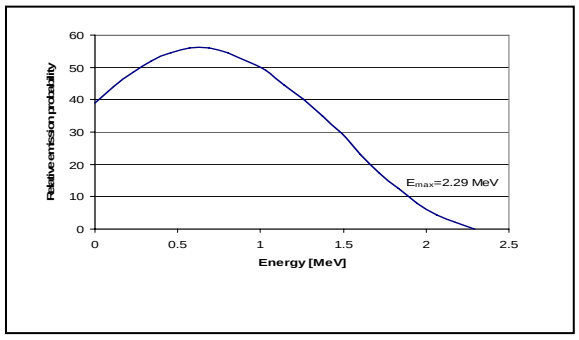

Fig. 4. Spectrum of photons from bremsstrahlung flying out of a 2 cm UO_2 sphere with density 10 g · cm^{-3}

Fig. 4 shows the spectrum of bremsstrahlung photons emitted from a 2 cm UO_2 sphere with a density of 10 g · cm^{-3}. The spectrum was tallied (f4 tally) in 10 keV intervals for a cell represented as a shielding box of 10x10x10 cm with a wall thickness 1 mm filled with air. Two significant peaks are K_α and K_β X-rays of uranium.

The bremsstrahlung spectra were calculated for the three source geometries as defined above (spherical with diameter 2, 5 and 9 cm). These were then used as input spectra for tallying the detector response for bare and shielded sources. The numbers of the bremsstrahlung photons which are escaping from the 2, 5 and 9 cm UO_2 sources as defined above are 0.0326, 0.0272 and 0.0290 per one beta particle starting in the source, respectively.

3. Final shape of the spectra

The beta particles emitted from unshielded sources on the detector response are contributing directly as electrons depositing their energy in the sensitive volume of the detector, and they are producing bremsstrahlung in the materials covering the detector crystal as well. The contribution of beta particles reaching the detector is not very important, as can be seen from Fig. 5. It is usually absorbed by the shielding. Final detector responses for the seven gamma lines, the bremsstrahlung in the source and the beta particles from the source are displayed in Figs. 5 .

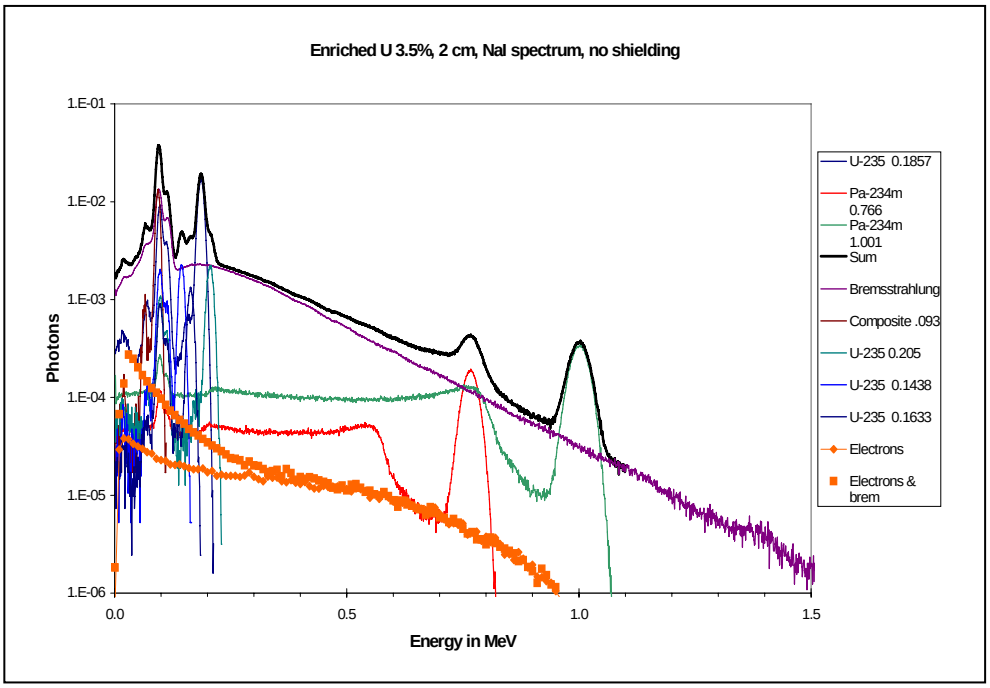

Fig. 5. Partial and final detector responses from 2 cm UO_2 source without shielding.

4. Investigation of an alternative approach

The bremsstrahlung produced by betas from ^{234m}Pa represents an important part of uranium spectra. It is changing the overall shape of the spectrum, and the uranium gamma and X-ray lines used for the analysis and detection are superimposed on this continuum. The mean of this continuum is approximately at 400 keV, so it is not so much shielded as the ^{235}U gammas at energies 186 and 205 keV (5 mm Pb will reduce photons at this energy ~ 4 times, at 205 keV more than 200 times, and at 186 keV more than 850 times). An example how the appearance of the Uranium spectra changes in the low energy with the lead shielding is shown in figure 6.

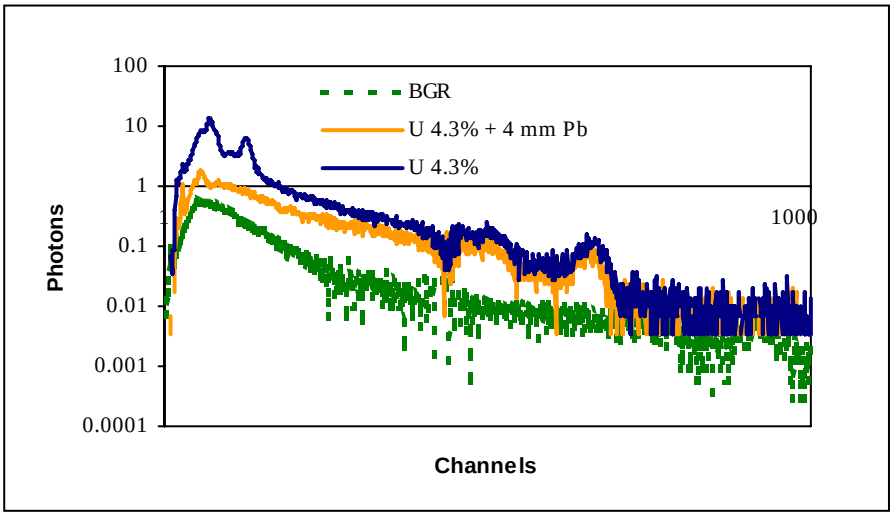

Fig. 12. Measured spectra with NaI(Tl) detector.

The most difficult case for detection is highly enriched uranium (90 % ^{235}U) shielded by lead (a lead thickness of 5 mm was assumed in the present studies), where the characteristic gamma rays from ^{235}U can no longer be detected. The dose rate in this case is really low, and it is almost only due to bremsstrahlung and high energy gammas from ^{234m}Pa. Evaluation of the spectrum in terms of integral counts within suitable energy intervals still will give a possibility to make a further analysis, if compared to the known responses of similar sources. Table 3 shows that it is not possible to detect a smaller amount of HEU (about 50 g) under shielding. However the ratio (signal + background/ background) is increasing with the uranium amount, and for ^{235}U masses above about 300 g the detection of highly enriched uranium becomes possible using this new approach.

Table 3. Sensitivity analysis for Highly Enriched Uranium for Fieldspec NaI(Tl) detector 1.2x1.5"

	Shielded by1 mm of lead				**Shielded by 5 mm of lead**			
Energy interval (keV)	400-600	400-1100	300-600	300-1100	400-600	400-1100	300-600	300-1100
	Counting rate per sec							
Background	5.58	10.90	12.74	18.06	5.58	10.90	12.74	18.06
42 g HEU	0.84	1.59	1.55	2.30	0.56	1.12	0.93	1.49
327 g HEU	5.78	11.08	10.52	15.83	3.86	7.85	6.35	10.34
1000 g HEU	18.92	36.16	34.61	51.85	13.13	26.78	21.46	35.12
	(Signal + Background)/Background							
42 g HEU	1.15	1.15	1.12	1.13	1.10	1.10	1.07	1.08
327 g HEU	2.04	2.02	1.83	1.88	1.69	1.72	1.50	1.57
1000 g HEU	4.39	4.32	3.72	3.87	3.35	3.46	2.68	2.94

5. Conclusion

The MCNP simulations were carried out (i) to model the emission characteristics of relevant specific gamma lines and of the bremsstrahlung emitted from different geometries of sources of and HEU, LEU, NU and DU materials under various gamma shielding, and (ii) to model the characteristic gamma responses of typical scintillation detectors (NaI) as used in hand-held monitors. The limits of the identification of uranium under shielding by gamma spectrometry are then defined. The results of this study will be used for testing the performance of any hand-held device in the identification of any uranium materials. From a review of the spectral data we are proposing a new approach for the identification of uranium which makes also use of the shape of the measured gamma continuum observed from shielded uranium. This complementary information can be helpful in cases of poor statistical data for the full photon energy peaks of uranium gamma rays. Further investigations of this approach are still necessary, especially for uranium mixed with other radioactive sources.

Session 6

New Threats and developments

Safeguards and Security Interface – Consequences for Instrumentation

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Abstract:

The threat of nuclear and radiological terrorism has inspired a multitude of initiatives to develop and improve technological and institutional capabilities to prevent the proliferation and illicit trafficking of nuclear or radiological devices, materials, and technologies. Due to the nature of misuse and diversion scenarios, terrorism threats are similar to some of the challenges encountered by the international non-proliferation community. Therefore, authorities with counter-terrorism responsibilities often turn towards the Safeguards community to take advantage of methods and instrumentation that have proven effective for verification of non-proliferation commitments. Such methods include material control and accountability, as well as – in the context of the implementation of the Additional Protocol (INFCIRC /540 corrected) – the detection of undeclared materials and activities. The following paper will identify instrumentation suitable for application in counter-terrorism scenarios, such as Safeguards specific non-destructive assay (NDA) and containment and surveillance (C/S) equipment. Next, the paper will outline challenges to adapt such Safeguards instrumentation for security enhancing applications. In this process of adaptation, the paper will then highlight the advantages of networking different equipment categories to provide more comprehensive barriers and protection against diversion and misuse of materials. A list of preliminary recommendations for implementation of such comprehensive approaches concludes the paper.

Keywords: trafficking, safeguards, material control, accountability

1. Introduction

The threat of terrorist actions employing nuclear or radiological devices has caused numerous changes to existing public security policy as well as the formulation of new initiatives intended to strengthen borders and infrastructure against the trafficking of nuclear or radiological devices, materials, and technologies. In addition to policy changes, new technologies and methods of operation have been developed to respond to terrorist threats.

Despite the deeply rooted differences in purpose and application, some aspects of Security activities are similar in their goals to Safeguards activities when it comes to the usage of technical instrumentation. Authorities in both arenas seek to prevent the spread of restricted materials and to detect diversion in a timely manner. The fact that counter-terrorism calls for immediate responses

while Safeguards goals are subject to the International Atomic Energy Agency (IAEA) timeliness strategy matters little for the detection technologies used. Similarities extend to specific conditions and responsibilities within both communities, including the need to monitor material trafficking areas and to secure restricted materials in a variety of forms. In addition, comprehensive monitoring systems in use in both counter-terrorism and Safeguards activities must integrate a variety of instrumentation: In counter-terrorism systems this may include biometrics, surveillance cameras, and radiation, metal and explosives detectors, while Safeguards systems may include cameras, radiation detectors, and electronic container seals.

As a result of these similarities, solutions developed specifically for use within the Safeguards community can under certain circumstances be adapted to meet the needs of Security authorities. These include specific technologies and systems developed within the private sector as well as general approaches implemented at the government level. On the other hand, there also exist fundamental differences between both areas that reflect on the potential of Safeguards instrumentation to meet counter-terrorism goals and vice versa.

This paper will identify interface points between Safeguards and Security efforts (with a focus on counter-terrorism) and will summarize programs and technologies currently in use in the Safeguards arena appropriate for use in counter-terrorism scenarios, including non-destructive assay (NDA) equipment and customized monitoring and surveillance systems. Some solutions need to be adapted for application in the counter-terrorism field. The advantages and challenges of adapting these programs and technologies will also be illustrated. An overview of expanding cross-over solutions between Safeguards and Security applications concludes the paper.

2. Safeguards and Non-proliferation Technologies

Safeguards approved technologies found use in Security environments even before the extensive burst in counter-terrorism technology development spurred by the September 11, 2001 attacks. The Department of Energy (DOE) Material Protection Control and Accountability (MPC&A) initiative has been employing Safeguards approved surveillance instrumentation in its MPC&A Operations Monitoring (MOM) program since early in 2001. This instrumentation is not used to safeguard existing materials, but rather to provide an additive layer of security by monitoring personnel, materials, and procedures at Russian nuclear facilities. In this application, Safeguards equipment also serves as a management tool to verify that personnel are working in accordance with security procedures and that security upgrades are properly installed and operated.

After the need for Homeland Security and counter-terrorism took on a new dimension (most notably in the United States in 2001), attention turned to quickly finding accomplished solutions to enhance security and prevent further terrorist attacks. Although by definition Safeguards has only little to do with physical protection or security, it was one area investigated as a possible source of adaptable technologies. The Safeguards mission is to detect diversion of nuclear materials in a timely manner and does not include the physical protection of such materials. However, because the monitoring and detection capabilities developed for Safeguards efforts bear certain similarities to those applications needed in counter-terrorism efforts, Safeguards instrumentation was promoted for security enhancements. The areas of applicability for Safeguards instrumentation in counter-terrorism activities include detecting illicit trafficking of nuclear or radiological materials, video surveillance, and tamper-indicating sealing of containers to prevent smuggling of illicit cargo.

The following lists Safeguards approved systems which either have the potential to be adapted or have been adapted to meet the goals of the counter-terrorism community.

2.1 Vilnius Airport Monitoring System

Under the DOE's Second Line of Defense (SLD) Program, a monitoring system was installed at the Vilnius Airport in Lithuania to enable local and international authorities to detect and counter the illicit trafficking of both radiological and nuclear materials. SLD serves as an additive layer of detection and security capabilities to prevent stolen materials from leaving their host country, thereby supplementing the MPC&A initiative (which can be interpreted as the First Line of Defense).

The monitoring system at Vilnius employs a system of gamma-neutron radiation detection portal monitors deployed at key passenger and cargo chokepoints, with Safeguards approved cameras positioned to monitor each portal. If a portal monitor is triggered, cameras start recording pictures, an alarm alerts airport security personnel, and alarm data (images and detector reading protocol) are shared between airport security headquarters, national border security, and the US DOE. The review of this data is facilitated by standard IAEA data review software.

A system such as the one currently employed at Vilnius could be supplemented with additional instrumentation (e.g. biometrics devices, explosive and chemical detectors, and additional radiation sensors or cameras) in order to broaden its applicability in counter-terrorism efforts. The main challenges in this adaptation process consist of integrating additional equipment into the system and combining the data generated by various sensors into a cohesive picture of alarm events. A more detailed description of the adaptation of an integrated safeguards system in border security is presented by Zack and Ondrik [1].

2.2 Non-Destructive Assay (NDA) Equipment

In Safeguards, NDA equipment is typically used by IAEA inspectors to verify and confirm the presence of declared safeguards relevant materials at declared nuclear facilities. Another well known application of NDA is the identification of large quantities of nuclear or radioactive waste before storage. While both applications assume the detection of an expected amount of a known material with a known signature, counter-terrorism applications are required to detect the presence of an as-yet unknown material in unspecified amounts. Because the use of NDA equipment in counter-terrorism activities is inherently different, it is usually approached in a two-step process: First, a primary screener (gamma and/or neutron based) identifies significant increases in the radiation level and alerts security personnel. The source triggering the primary screener is then identified in a secondary screening process.

Although the objectives differ between Safeguards and counter-terrorism efforts, the NDA technologies (both hardware and software) utilized remain essentially unchanged; only the human interface differs. The same detectors that detect and identify the signature of the gamma radiation and the same He^3 tubes used in neutron detection can be applied towards Safeguards or towards counter-terrorism. Similar parallels exist for the calculation algorithms. In this respect, the technologies that were developed for Safeguards have quickly found applicability in anti-terror activities. There are, however, additional obstacles in the adaptation process, which will be discussed in Section 3 below.

2.3 Next Generation Surveillance System (NGSS)

Slated to serve as the IAEA's standard surveillance system for the next ten years, NGSS will replace the IAEA's current digital surveillance system (DCM-14) and will be installed at nuclear facilities in order to monitor activities involving declared nuclear material. The development of NGSS will include some advanced features that will broaden the applicability of such Safeguards instrumentation towards use in counter-terrorism efforts. In its basic configuration NGSS will consist of a variable number of sensors linked to an individual data management core component. This flexibility will allow NGSS to serve as a platform for any sensor with an interpretable data output. NGSS will be capable of integrating sensors from a variety of disciplines, including video and NDA equipment, as well as counter-terrorism-relevant sensors such as explosives detectors, x-ray devices, and biometric scanners. The cross-triggering capability of networked sensors and the integration of diverse sensor data streams to provide a more complete overview of relevant events will also promote NGSS for counter-terrorism use.

NGSS is currently planned for deployment by the IAEA in declared reactors, waste depositories, and other facilities which store or utilize nuclear materials. Although it will be used by the IAEA primarily in the monitoring of declared nuclear facilities in member states, the core concept of NGSS promises to be appropriate for installation at sites other than authorized nuclear facilities.

3. Adaptation Challenges

Safeguards instrumentation is usually developed in custom research and development projects and is designed in accordance with IAEA user requirements. While commercial-off-the-shelf (COTS) components are used as much as possible in order to reduce production costs, the custom development of some components is required because of their highly specific requirements. The resulting instrumentation is so closely tied to its Safeguards application that conversion to an application outside its core field of use can pose significant problems.

First, the development of custom designed components of Safeguards instrumentation is very costly. Should Safeguards-specific components of such instrumentation lack applicability in counter-terrorism, it is therefore attractive to replace them with cheaper equivalent components from the consumer market. Expensive custom components include data authentication devices and tamper-indicating equipment enclosures. While consumer products are available with features similar to Safeguards video surveillance system without these expensive features, closer examination shows that both tamper-indication and authentication have direct applicability towards counter-terrorism goals.

In Safeguards data authentication is employed to ensure that data generated at nuclear facilities between inspection visits are not tampered with. At first glance, data authentication seems unnecessary in counter-terrorism applications (e.g. border crossing control, airports, etc.) because security personnel are present at all times. However, authentication has security applicability in that data generated during an alarm event can later be used in prosecution and can serve as evidence in a court of law. If legal procedures are anticipated during the design of the security system, authentication of data should be a mandatory requirement.

Because nuclear facilities safeguarded by the IAEA are under the control of their respective host countries, the Agency defines these installations as potentially hostile environments (in the event that a country decides to divert materials without detection). Tamper-indicating enclosures ensure that instrumentation generating and storing Safeguards relevant data is not tampered with between inspection visits. Once again, the constant supervision by security personnel makes expensive tamper indicating features appear unnecessary. However, if terrorist conspiracies involving insider and/or collusion threats are considered, security personnel should not be granted undetected access to instrumentation at sites such as border crossings, or other ports of entry. As Kadner et al [2] note, “(an) intrinsic independence from operator control impedes and deters from acts of sabotage and theft, thus contributing to the overall secure containment of all radiological materials and the prevention of illicit trafficking of such materials.”

The principal in adapting Safeguards technologies for counter-terrorism applications lies in the interaction of the user with the equipment. Safeguards inspectors are highly trained individuals specializing in the analysis of Safeguards relevant data and are proficient in the use and troubleshooting of Safeguards instrumentation. Security personnel generally do not receive a comprehensive level of training and must be proficient with sensors from a variety of disciplines (biometrics, chemical sensors, x-ray devices, etc.) as well as radiation sensors. Although the technology accomplishes identical goals in both applications, the user interface must be adapted to accommodate differing user clienteles. This applies less so for Safeguards surveillance systems which automatically generate alarm information than for NDA detection equipment.

While the basic detection hardware employed in NDA equipment remains the same in counter-terrorism applications as it does in Safeguards, the user software and detection parameters must be modified to reflect the differences in goals. In addition to this principal adaptation challenge, some repackaging efforts are required to make Safeguards-specific NDA equipment suitable for use in counter-terrorism applications. NDA equipment as utilized in Safeguards employs a number of features which are not needed in counter-Terrorism applications and might only confuse security personnel that are not nearly as trained as an IAEA inspector. And, while Safeguards personnel must be able to modify a variety of factors and parameters of NDA equipment, security personnel are rarely required to do so, making the detailed software interfaces of Safeguards equipment superfluous and cumbersome in counter-terrorism applications.

The re-design and adaptation of NDA equipment for use in counter-terrorism applications should ideally follow the example of surveillance systems which call for complete automation of the detection process. The exposure of security personnel to the complex algorithms used in detection equipment is kept to a minimum in an automated system. While a limited user interface is still necessary to inform

the user of equipment status and to assist in troubleshooting, minimizing user access to the core functions mitigates the chance of personnel improperly handling the equipment.

Some of the above recommendations have already been implemented in the Vilnius airport security system. Radiation detection portals automatically sample the background and scan passengers and cargo. In case an alarm is detected, authorities are immediately notified and the cameras integrated into the system automatically start recording images. Alarm data is transferred to a review station where it can be used to coordinate response efforts and to verify proper response behavior by security forces.

4. Conclusion

The application of Safeguards instrumentation in counter-terrorism scenarios is interesting in principle, but is fraught with challenges due to difference in goals and in the user training required to operate the resulting technologies. Implementing Safeguards equipment represents a more stable approach than total reliance on overworked inspectors and intelligence-gathering efforts [3]. The more Safeguards system components can be integrated into a single detection system and the further alarm data acquisition processes can be automated, the easier it is to convert Safeguards instrumentation for use in counter-terrorism. Because counter-terrorism efforts involve a variety of sensor technologies, system integration in Safeguards should receive more attention and should be expanded to include instrumentation beyond those traditionally employed in Safeguards applications.

The concept behind the NGSS (a uniform modular data acquisition and management component connectable to any sensor type and capable of being integrated into a comprehensive, cross-triggering sensor network complete with data review and archiving) is one approach that promises to facilitate that. Safeguards development efforts may once again provide innovative technologies with possible application in the security and counter-terrorism field.

The potential for Homeland Security and counter-terrorism to provide innovative technologies with application in Safeguards increases as the discipline matures and becomes a significant field of innovative technology development. In particular, the extensive use of integrated safeguards and the implementation of the Additional Protocol[1] mean that counter-terrorism solutions in the area of radiation detection will be useful for inspectors, who will be required to visit undeclared sites with minimal prior notice. IAEA inspectors will face a lack of knowledge regarding the nature or quality of materials present as well as a lack of knowledge of what (if any) undeclared activities have taken place at a facility. This situation will be similar to counter-terrorism situations, wherein passengers and cargo of unknown origin and content are screened. In addition, the need for portable detection equipment (currently used in counter-terrorism scenarios during secondary screening) might find use by IAEA inspectors visiting undeclared facilities.

In light of the above, it is recommended that both Safeguards and counter-terrorism experts maintain an open dialogue to investigate and facilitate the greatest possible synergies between both fields and that they support one another in their pursuits to the greatest extent possible.

References

[1] Zack, N., Ondrik, M.; *Augmenting Security at Airports, Seaports, Reactors, Railways, Borders, and Private Facilities*; Proceedings of the 43rd Annual Meeting of the Institute of Nuclear Materials Management, 2003.

[2] Kadner, S., Stein, M., Zack, N. Butler, R., Gallimore, T.; *Homeland Security Applications of Safeguards Equipment*; Proceedings of the 44th Annual Meeting of the Institute of Nuclear Materials Management, 2003.

[1] INFCIRC /540 (corrected)

[3] Kadner, S., Doyle, W., Kraus, S.; *Applications of Nuclear Safeguards Technologies: Border Security Initiatives*; Proceedings of the 43rd Annual Meeting *of the Institute of Nuclear Materials Management,* 2002.

Techniques for locating and identifying nuclear and radioactive material

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Abstract:

Reliable measurement techniques are imperative for early detection of illicit use of nuclear and radioactive material. Nuclear safeguards are an important tool to impede the illicit use of fissionable material. But the scope of safeguards is only nuclear material. Other radioactive substances which are widely used in industry and medicine are not controlled to the same extend. Terrorist attacks including this type of material may have far extending consequences. Therefore it is of great importance that mobile measurement systems are available to detect and to respond to malicious acts involving radioactive or nuclear material.

Fraunhofer INT has built up a mobile measurement system in a transportable container for the detection and identification of radioactive and nuclear material. The system is equipped with various types of detectors for neutron and gamma radiation, some of the detectors are similar to those used in safeguards. The measuring capabilities of the system have been extended by a gamma scanner. The scanner has two interchangeable detectors. A CsI crystal with relatively high efficiency and a CdZnTe-room-temperature semiconductor with high resolution. The area of interest is scanned sequentially and for each measuring point a standard gamma spectrum is recorded. Regions of Interest can be used to select specific energy ranges. A digital optical image of the measurement area is overlaid with the color coded result of the gamma measurements.

The gamma camera has been used to locate and identify radioactive sources with and without shielding material. Results of these measurements will be presented.

Keywords: illicit trafficking; in-situ measurements; nuclear terrorism; dirty bomb

1. Introduction

The threat of terrorist attacks has been a major concern especially in the past years. Several measures have been discussed and fully or partly implemented to address this threat. Very important in this context is the control of nuclear material which is covered by the safeguards regime under the non-proliferation treaty. The efforts towards strengthened safeguards under the additional protocol allow for verification of the absence of illegal nuclear activities in addition to the verification of the nuclear material in the fuel cycle itself.

On the other hand not only nuclear material is of potential interest to a terrorist. Other radioactive material, for example, sealed sources with high activity which are widely used in industry, are also of concern. The surveillance of such sources in general is not as complete as the surveillance of the material in the nuclear fuel cycle. In most countries such sources are controlled by the radioprotection authorities. But in some less developed countries there are no or insufficient radiation protection regulations or even the radioprotection authority is missing. This material as well as nuclear material may pose a threat when used in a dirty bomb.

2. Techniques

Various techniques can be used to detect, locate and identify nuclear and radioactive

material by means of radiation detection.
If the suspicion of a threat with nuclear or radioactive material arises the first task is to detect the presence of such material. This may be done by a survey of a particular region by a helicopter equipped with suitable detection systems or by a measurement car. Detection from the air requires a rather large activity of the source.

After artificial radiation is detected by the measurement system in the car or helicopter, in the second step the suspicious object is located by means of radiation search tools. Hand search systems based on Geiger-Mueller detectors or NaI or CsI scintillators are often used for this task. A person is required to get close to the object to do the search if no robot is applied.

The third step is to identify the material for the preparation of a risk assessment and delaboration. Identification normally requires high resolution detectors like high purity germanium (HPGe) detectors. For the identification it is normally also necessary to get very close to the object.

3. Mobile detection system

Fraunhofer-INT has built up a measurement system for the detection and identification of nuclear and radioactive material which is integrated into a transportable container [1]. The container is shown in figure 1. This system is completed by a car equipped with measurement systems to detect and locate the material.

Figure 1: Transportable container, gamma and neutron measuring devices. On the right side power generator on separate trailer

The system is now further completed by a gamma scanner. The advantage of a gamma scanner is that during the scan nobody needs to stay in the vicinity of the hazardous object.

4. The gamma scanner

4.1. System design

The gamma scanner [2] consists of two units, the measuring head itself and a portable control unit. Both components are connected with a cable of up to 80 m length. The power supply of the whole system is placed inside the control unit. Only one hermetic watertight cable, which is easy to decontaminate, is needed for connecting the control unit with the measuring head. To set up the instrument there are only 3 parts which need to be carried, the control unit, the measuring head and the cable drum. Each part has a weight between 10 and 20 kg, therefore the whole system can easily be moved. A picture of the measuring head of the RoSCAN is shown in figure 2.

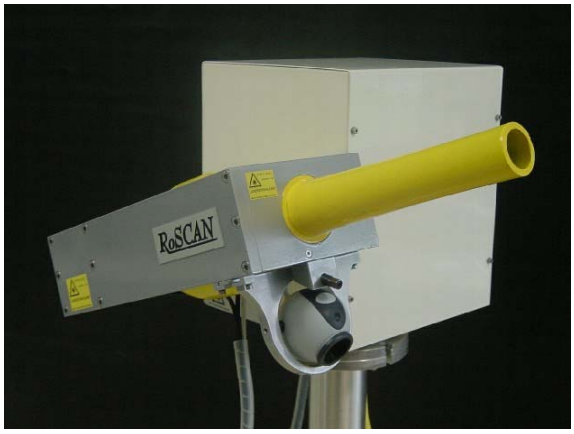

Figure 2: Measuring head of the RoSCAN

The measuring head itself consists of a robust metal case which includes the stepper motors, a digital camera, the collimated detector, a multi channel analyzer MCA166, a laser pointer and the electronic control board. The digital camera for the image in the visible spectrum works with a resolution of 640x480 pixel. Two different detectors with their own collimator can be used to measure the gamma radiation: a CsI detector, which provides high efficiency but limited energy resolution and a CdZnTe (CZT) detector, which provides high energy resolution

but low efficiency. The laser pointer is used to mark active fields which have been identified by the gamma measurement.

The control unit which is placed in a briefcase contains the power supply, a communication module for the communication with the measuring head, and a laptop.

All functions of the measuring head, including the camera and the MCA-166 are controlled by a computer program which is named RoSCAN according to the system name.

The optical picture of the scanned area is subdivided into rows and columns in the form of a rectangular matrix. At each field of the matrix a gamma measurement is acquired for a predefined time and a complete gamma spectrum is recorded. The scan starts at the upper left corner of the measurement area and then proceeds to the right until the right border is reached. The energy spectrum of each matrix point can be evaluated separately or the sum spectrum of the whole measurement area can be processed.

In order to get an overview picture of the gamma scan the intensity of each single measurement is converted into colors and used to build a color map superimposed on the optical picture. Two regions of interest can be set in the spectra for this purpose. Each of these two ROIs can be used with the colors representing either the integral count of the ROI or the peak area of the ROI.

In addition an extra measurement can be performed at any single field of the matrix. For example, a hot spot found in the scan can be investigated in more detail.

The opening angle of the collimated gamma detector is 5°. The practical usable matrix has a minimal dimension of 6x8. The maximum matrix resolution is 36x48, with interpolation 108x144. Interpolation means that any field is divided into nine fields and these smaller fields are filled with interpolated values. The rotating angle of the device is 360° in horizontal direction and 220° in vertical direction.

The CsI detector is a cylinder with a diameter of 25 mm and a thickness of 25 mm. The usable energy range extends from 100 keV to 3000 keV and the energy resolution of the detection system with this detector is about 10 %, that means a FWHM of 60 keV for the 662 keV line of Cs-137. The light of the CsI crystal is collected by a photo diode. The collimator is made of lead with a thickness of 38 mm. No high voltage is necessary for this detector.

The CZT detector has an active volume of 500 mm^3 with quasi hemispheric shape. The diameter of the crystal is about 10 mm. Its energy range is from about 30 keV to 2000 keV and the energy resolution is 15.9 keV for the 662 keV line of Cs-137. The operating voltage is 1500 V. Again the collimator is made of lead with a thickness of 38 mm. As the active diameter of the CZT detector is much smaller than the diameter of the CsI detector the collimator is also much shorter.

The volume of the CsI crystal is 25 times bigger than that of the CZT crystal and the detection efficiency of the CsI is approximately 20 times higher for energies in the range of Cs-137 and approximately 30 times higher for energies in the range of Co-60.

4.2. Gamma scanner results

Calibration measurements with Cs-137 and Co-60 have been performed [3]. Typical spectra are shown in figure 3 and figure 4.

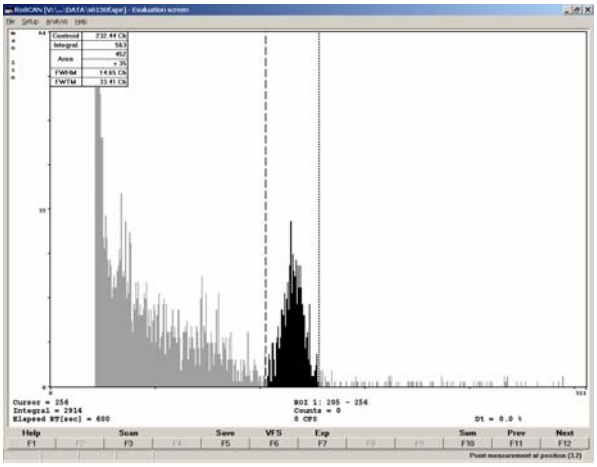

Figure 3: Energy spectrum of a Cs-137 source, measured with the CsI detector

A set of measurements was carried out in which the two sources were placed in a suitcase simultaneously. A Co-60 source with an activity of 2.1 MBq was placed on the left side and a Cs-137 source with an activity of 288 kBq was placed in the bottom right corner. The suitcase was placed in a distance of 100 cm, 150 cm and 200 cm from the measuring head of the RoSCAN. Figure 5 and figure 6

show that the position of the sources easily can be retrieved from the measurements.

Other measurements were performed with different types and thicknesses of shielding material.

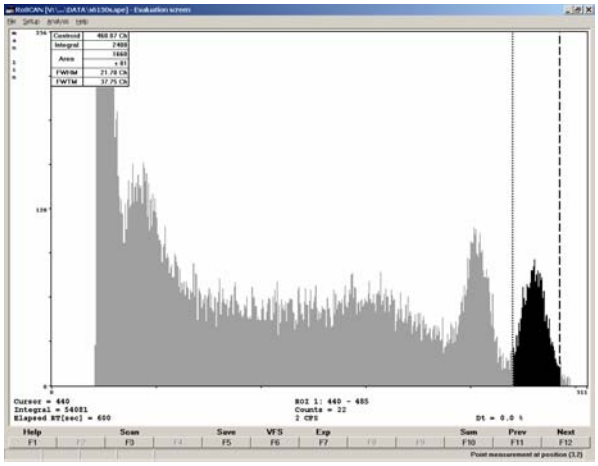

Figure 4: Energy spectrum of a Co-60 source, measured with the CsI detector

We also performed measurements with the system outside of the laboratory with distances of the ROSCAN head to the radioactive sources of up to 33 m. This requires sources with some 100 MBq and long measuring times. In the experiment a 270 MBq Cs-137 source was scanned and located with a measuring time of 60 s in a distance of 13 m but this activity was not sufficient to locate the source in a distance of 33 m.

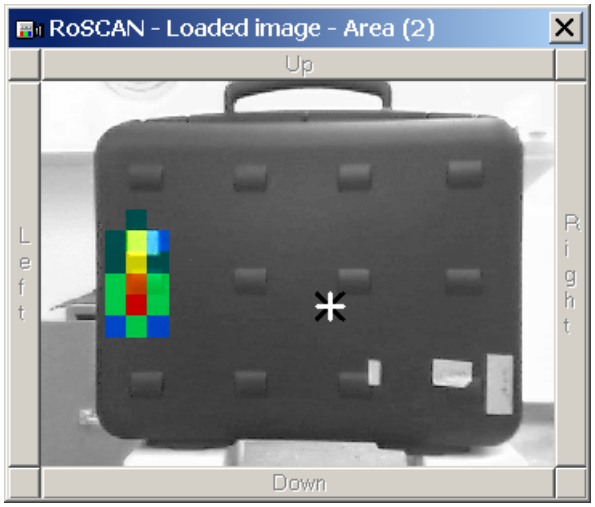

Figure 5: Scan of a suitcase, grid 6x8, a Co-60 source and a Cs-137 source were placed inside the suitcase. The colors represent the peak area of the 1173 keV line of Co-60 with interpolation between neighboring measurement points. Distance 100 cm, measuring time 100 s. (The grey areas are stripes of adhesive tape attached to the suitcase)

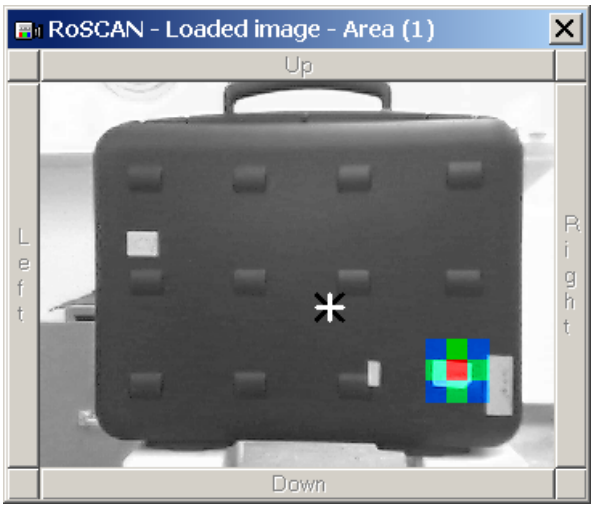

Figure 6: Same as figure 5 except that the colors represent the peak area of the 662 keV line of Cs-137.

5. Conclusion

The transportable detection system for radioactive and nuclear material of INT has been complemented by a gamma scanner. The gamma scanner allows to remotely scan for radioactive material. The measurements have shown that the device is able to locate radioactive objects in reasonable time. Gamma scanning can help to accomplish the task of unattended locating of radioactive material for instance in the case of other hazardous material present.

Consequently gamma scanning may help to prohibit nuclear terrorism

Acknowledgement

The authors thank Mrs. Sütterle for performing parts of the evaluation of the measurement.

References

[1] Rosenstock, W.; Köble, T.; Risse, M.; *Measurement techniques to combat nuclear terrorism*; Proceedings of the 25st annual meeting symposium on safeguards and nuclear material management; Stockholm; 2003; 465

[2] Keubler, J.; *RoSCAN user manual*; Rossendorf, Germany; 2003

[3] Köble, T.; Rosenstock, W.; Risse, M.; *Locating nuclear material by gamma scanning;* 45th Annual Meeting of the Institute for Material Management (INMM); Orlando, Florida, USA; July 2004; paper 0192

IAEA Activities in Supporting Member States to Combat Illicit Trafficking

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Abstract:

Nuclear terrorism poses four types of nuclear risks: Nuclear weapons acquired by theft, nuclear explosive devices created from stolen nuclear material, radioactive dispersal devices (RDDs) – dirty bombs and radioactive hazards caused by an attack on, or sabotage of, a facility or transport containing nuclear or radioactive materials. A strategy to counter nuclear terrorism is founded on measures to prevent thefts of nuclear and other radioactive materials (first line of defense) and to provide security in the case that the prevention was not successful (second line of defense). In the paper we explain briefly the additional programmatic effort in the latter, which the IAEA had undertaken, on request of Member States, after 2001. The support to countries in the field of detection of illicit trafficking by radiation monitoring at borders and in a country (evaluation missions, provision of equipment and training) is described in more detail.

Keywords: Nuclear security; border monitoring; illicit trafficking; radiation; nuclear security, nuclear terrorism

1. Introduction

After 9/11 concern about the security of nuclear and the radioactive material has received an increased level of attention. Analysis showed that there is a threat that sub national terrorist groups may be planning attacks, using a combination of classic explosive and a radioactive substance – a radiation dispersal device (RDD) or dirty bomb. Since some terrorists are willing to loose their lives, the earlier assumed hypothesis of the self-protection of strong radioactive sources may not be valid and security measures may need to be revised. Although less likely, the possibility that sub nationals groups would acquire a nuclear weapon must also be taken into consideration.
In September 2001, the General Conference of the IAEA requested a review of Agency's activities relevant to preventing nuclear terrorism and proposals for strengthening measures to combat nuclear terrorism. In this review, the following four threats of nuclear terrorism were identified:

- Theft of a nuclear weapon
- Theft of nuclear material to make an improvised nuclear explosive device
- Theft of other radioactive material for malicious acts, including a RDD
- Sabotage of a nuclear facility or a transport.

2. IAEA Nuclear Security Plan of Activities

Reacting to the request of the General Conference, the IAEA established a "Nuclear Security Plan of Activities", which was adopted by the Board of Governors in March 2002. That plan will conclude at the end of 2005 and a new plan, based on the lessons learned from the plan's three years of operation

is being developed. In addition a Nuclear Security (Multi - Donor) Fund (NSF) was set up calling upon Member States to make extra budgetary, voluntary contributions, which would allow the implementation of the this programme. It was estimated that approximately $11 million per year would be needed. The Board established a Nuclear Security Fund for this purpose utilizing voluntary contributions by Member States. As of April 2005, 24 Member States and one non-governmental organization had pledged a total of about $38 mio. The comprehensive plan is organized around eight activity areas described briefly below:

Activity Area I: Improving physical protection of nuclear material and facilities

These measures comprise the first line of defense – the prevention of the unauthorized removal of nuclear and other radioactive material from a facility and the prevention of an attack on a nuclear facility or source transport. The "International Physical Protection Advisory Service" (IPPAS) assists Member States to strengthen and enhance the effectiveness of their physical protection systems to secure nuclear materials and facilities. Based on the recommendations of the IPPAS report (covering, e.g., legislation, licensing and regulations, facility implementation, inspections and enforcement) technical support, e.g., upgrading of facilities may be provided.

Figures 1a and 1 b: A secure fence around a sensitive object is a key physical protection requirement

Activity Area II: Detecting malicious activities involving nuclear and other radioactive materials

Physical protection measures constitute a second line of defense and come into play when nuclear material or a radioactive source had penetrated the first line of defense. In the early 90's, stolen and smuggled material was primarily confiscated through intelligence. This situation has changed after the installation of border monitoring equipment at airports, land and sea borders. In Russia, e.g., presently more than 95% of the seizures are detected through technical means – several 100s of portal monitors covering more than 50% of the border crossing points [1]. The activities of area II include advisory services, provision of equipment, training and are discussed in detail below. It should be

noted that last year the UN Security Council adopted Resolution 1540, which requires all amongst other States to: “Develop and maintain appropriate effective border controls and law enforcement efforts to detect, deter, prevent and combat, including through international cooperation when necessary, the illicit trafficking and brokering in such items in accordance with their national legal authorities and legislation and consistent with international law”.

Figures 1a (left) and 1b (right): Large scale installation of Radiation Portal Monitors in Russia (left, pedestrian/luggage monitors at an airport) and in the USA (right, vehicle monitors at a land border crossing point; [2]) credit N. Kravchenko and R. Kouzes

Activity Area III: Strengthening of the State Systems of Accountancy and Control (SSAC)

This activity utilizes synergies between safeguards and security. The objective is to strengthen the State’s SSAC so that the likelihood that nuclear material can be diverted to a sub national group is reduced. Measures include assessment missions (IAEA SSAC Advisory Service), seminars, workshops, development of SSAC guidelines, provision of software and equipment.

Activity Area IV: Strengthening of the security of radioactive materials other than nuclear material

Before the year 2001 the main concern with lost radioactive sources was connected with the possibility that they would show up in scrap or elsewhere, exposing people to potentially dangerous radiation exposures. After 09/11 an additional concern was identified – the possible use of radioactive sources for malicious acts, including a RDD. Consequently the security of radioactive sources places enhanced emphasis on inventory records, secure storage, use and transportation of strong radioactive sources in a country and the recovery and safe keeping of orphan sources.

Figures 3a (left) and 3b (right): Radiation monitoring at a scrap yard and radioactive item in scrap with about half mSv/h at the surface; credit P. de Bruin

Activity Area V: Assessing the security/vulnerability of nuclear facilities

This activity area is closely related to Activity Area I and is used to determine the vulnerabilities and threats to nuclear facilities. A Design Basis Threat (DBT) methodology is used to assist States in determining the possible threats to nuclear facilities in their country. A DBT is a profile of the type, composition, and capabilities of an adversary. It is used as a basis for designing physical protection systems to defend against acts of radiological sabotage and to prevent the theft of nuclear material.

Activity Area VI: Responding to malicious acts and threats

It needs to be pointed out that there are two kinds of response procedures. One is the response-to-detection (often called tactical response), which refers to the action to be taken in case of a trafficking incident (e.g., call of experts, determination of the isotope, assessment of health hazard, etc.; [3]. For more details, see par. 3.2 and 3.3.
The other relates to emergency response procedures that should be in place in case there is a serious radiological release, e.g., caused by a RDD. [4]

Activity Area VII: Ensuring compliance with international agreements and guidelines

A set of international agreements and guidelines is available, the adherence to which is the basis for a successful security policy I
n a country (e.g., Convention of the Physical Protection of Nuclear Materials and Nuclear Facilities, Code of Conduct on the Safety and Security of Radioactive Sources). Expert missions are conducted to encourage states, where appropriate to ratify and adhere to the full range of international legal instruments.

Activity Area VIII: Coordinating information on nuclear security

This is a broad activity area which, inter alia, organizes international conferences and operates the Illicit Trafficking Data Base (ITDB).

International Conferences

- International Conference on Nuclear Security: Global Directions for the Future (London, 2005)
- International Conference on the Safety and Security of Radioactive Sources: Towards a Global System for the Continuous Control of Sources Throughout Their Life Cycle (Bordeaux, 2005)
- International Conference on Security of Radioactive Sources (Vienna, 2003)
- International Conference on Nuclear Forensics (Karlsruhe, 2002)
- International Conference on Measures to Prevent, Intercept and Respond to Illicit Uses of Nuclear Material and Radioactive Sources (Stockholm, 2001)

Database services - Illicit Trafficking Data Base

The ITDB was established in 1995. It contains confirmed reports of incidents involving illicit trafficking in nuclear and other radioactive materials from 1993. The scope of the incidents includes situations, which relate to the unauthorized acquisition, provision, possession, use, transfer or disposal of nuclear materials and other radioactive materials, whether intentional or unintentional and with or without crossing international borders, including unsuccessful or thwarted acts. The ITDB contains authoritative information confirmed by States supplemented by information on the trafficking incidents obtained from various media sources. As of May 2004, 75 States participate in the ITDB programme. Almost all European States participate in the ITDB programme.
For reference see: [http://www.iaea.org/NewsCenter/Features/RadSources/Fact_Figures.html]

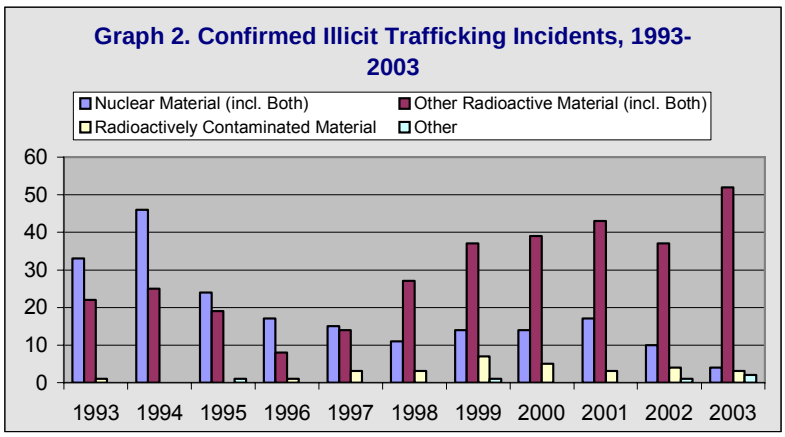


Figure 4: Statistical information on confirmed illicit trafficking incidents

3. IAEA Activities in Area II, Detection

With this as background, this paper will focus on one important aspect of the Agency's Nuclear Security Plan – radiation detection. Since humans cannot see or feel radiation, a key element for preventing illicit trafficking is the installation of radiation detectors at borders and in a country and the response to the detection of radioactive material. This comprehensive activity addresses assessment, provision of equipment, training, legal basis, responsibilities, detection and follow-up to a seizure.

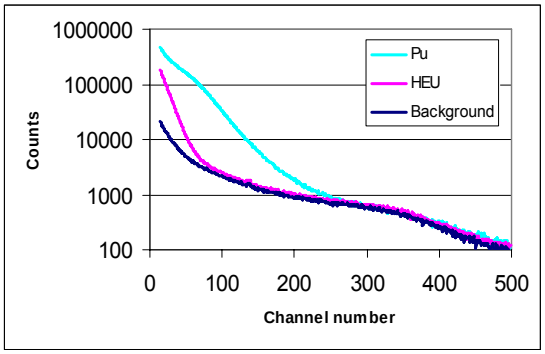
3.1. The evolution of automated radiation monitoring systems

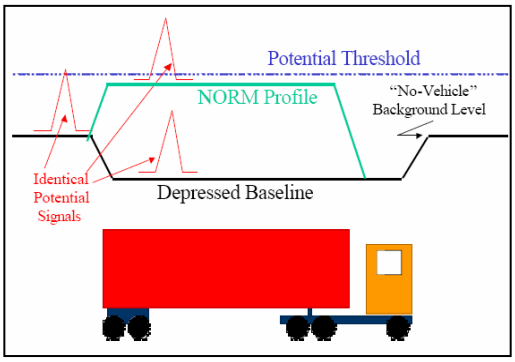
Automated radiation monitoring where persons or vehicles are channeled through a narrow gateway has a long history. The first systems were installed in nuclear facilities and power plants to detect radioactive material and contaminated items or people. They operated either indoors in a stop-and-go mode, allowing a sufficiently long measurement time to reach the needed sensitivity or outdoors in a drive – through mode. The catastrophic accident at reactor 4 of the Chernobyl nuclear power station in 1986 provided the incentive for the installation of automated radiation monitoring systems because of the wide spread contamination. These systems were the design prototypes for the present border monitors. They were built for outdoor use and employed either a large plastic scintillator or a large NaI detector for gamma detection. After the first incidents, where radioactive sources were melted in foundry operations, similar systems were installed at scrap yards and steel mills. These systems, however, did not use a neutron detector. The first border monitoring systems to detect illicit trafficking were installed in Russia, beginning in the mid90's. They were more sophisticated than the early systems, having additional features:

- Gross gamma and moderated neutron detectors
- Occupancy sensor, allowing background updating when not occupied
- Video subsystem, recording the monitored item or person.

Since then the development of radiation portal monitors (RPMs), has progressed further. Their large scale installation in the US after 9/11 provided an additional impetus to their further improvement. Modern systems have the following additional features [2]

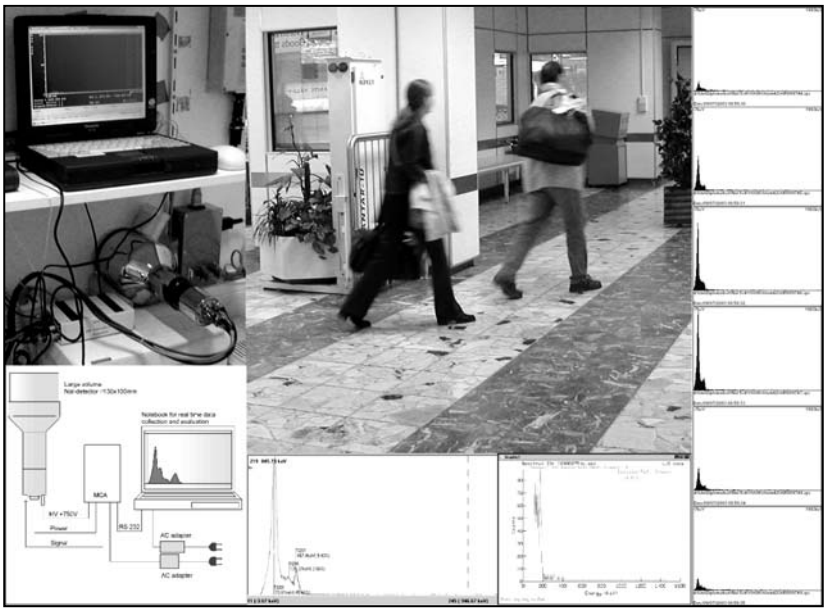
- Display of the neutron/gamma radiation intensity profile along the load of a truck on the computer screen, allowing assessment of the spatial distribution of the source. A point source would indicate an artificial source or a source distributed along the load may indicate NORM).
- Increased sensitivity by taking into account the suppression of the background due to the gamma absorption in the volume of a fully loaded truck (see figure 5b)
- Crude 4-8 window gamma "spectrometry" to assess the hardness of the gamma spectrum to differentiate between high energetic gamma emitters (NORM) and low energetic gamma emitters, e.g., nuclear materials (see figure 5a).
- Networking of different border monitors of an area or even a country to have real time information on radiation alarms at a central location.





Figures 5a (left) and 5b (right): The Compton gamma spectra of plastic scintillator allow to differentiate between nuclear materials and NORM (5a); The depression of the background reducing the sensitivity of a border monitor if not compensated (credit R. Kouzes)

The newest development is the introduction of spectral monitoring systems. This was prompted by the high number of so called "innocent" alarms – either caused by K-40, Th-232 or Ra-226 in truck loads or by radio pharmaceuticals in passengers at an airport [6]. Spectral border monitors allow automation of the present follow-up with a hand-held radioisotope identifier (RID), which can create an annoying delay for the public.



Figures 6a(left) and 6b(right): Automated spectrometric vehicle monitor tested at PNNL [6]; Early prototype of a automated spectrometric pedestrian/luggage monitor tested at the airport in Vienna [6].

According to international recommendations, fixed installed systems are complemented by hand-held radioisotope identification devices (RIDs) [7] to allow the front line officers to localize the source, to assess the dose rate and to identify the isotope and the significance of the alarm. RIDs may be used in conjunction with more sensitive neutron search detectors since the neutron detection sensitivity of the RIDs is often not sufficient to quickly find a neutron source in a vehicle, which had triggered the much more sensitive PRM. Smaller size personal radiation detector (PRDs) are recommended to be worn by officers at the belt. They can also be used to find a source and to widely assess its health hazard.

3.2. IAEA detection support to a country in combating illicit trafficking

After receiving a request from a Member State, the initial step is the evaluation of the situation in the country/region with respect to illicit trafficking. This is done by evaluation of open source information, consulting the Illicit Trafficking Data Base (ITDB) and by conducting an International Nuclear Security Advisory Service (INSServ) Mission.

During the Mission, international and IAEA experts in the field of nuclear law, illicit trafficking, radiation detection at borders and physical protection assess all aspects in a country, which are relevant to combating illicit trafficking, e.g.:

- The existence of a legal basis and a mechanism to enforce the law
- A clear definition of the responsibilities within the country (e.g., customs, border guards, ministry of health)
- The existence of procedures and arrangements to respond to the detection of radioactive material
- The availability of well functioning detection equipment
- Training at different levels, including instrument training of the end users
- The existence of an Mobile Expert Support Team (MEST), which can remotely and/or in the field provide expert detection support
- The existence of a laboratory capability and experts to investigate seized items to provide the legal evidence for a court case
- The existence of secure storage facilities for disused sources to safely and securely store confiscated radioactive material.
- Access to relevant information on trafficking incidents in the region, e.g., by joining the ITDB
- Arrangements to obtain the services of a regional nuclear forensic laboratory, which can as necessary, assist with the analysis of nuclear material to determine its origin, by matching the analysis results with a database of “fingerprints” of materials of different origin.

As a result of an INSServ mission an Integrated Nuclear Security Support Plan (INSSP) is established, which may recommend the provision of detection equipment for frontline officers and instruments for an expert team. In addition, training at different levels can be provided (e.g., awareness training for the decision makers, regional training courses, instrument training for the front line officers and trainers, which can multiply the knowledge). The funding comes from extra budgetary resources, the Nuclear Security Fund (NSF) or in some cases from the regular budget of the Department of Technical Cooperation. Alternatively the IAEA can seek bilateral support.

3.3. The Services of the Nuclear Security Equipment Laboratory (NSEL)

To provide better technology support to the crosscutting Nuclear Security Programme, early in 2003, the NSEL was established in the Division of Technical Support of the Department of Safeguards, Section for NDA Equipment and Seals. It consists of a laboratory, test sources, equipment and staff with experience in radiation detection at borders, using synergies between safeguards equipment and border monitoring technology. The staff of the NSEL has two major tasks – implementation and development support.
The implementation support covers the following activities:

- Procurement specifications for equipment purchased by the IAEA for donation to Member States according to the workplans established in the INSServ reports
- Technical evaluation of bids received
- Acceptance testing of all portable equipment, which is routed through IAEA HQs before being send to the recipient country; corrective action being made in case of equipment failure
- Acceptance testing of PRMs after installation by the vendor in the field
- Devising maintenance and preventive maintenance schemes
- Maintaining a supply of equipment for IAEA training courses
- Participation as technical experts in INSServ missions and instructors in training courses

The principle of performing acceptance tests at the IAEA before deployment of portable and transportable equipment is based on IAEA experience with safeguards equipment. Defective equipment is returned for repair to the vendor before shipment to the field where diagnostic, testing

and return is more difficult. The performance of acceptance tests by IAEA staff also provides a good understanding on performance limits and problems, which helps to improve the equipment gradually through feedback with vendors. In 2004, e.g., more than 230 portable equipment items were acceptance tested. The highest failure rate of up to 20% was observed for the more sophisticated RIDs.

In addition to equipment for front line officers, IAEA has also begun providing portable equipment for the Mobile Expert Support Teams. These are groups of experts, knowledgeable in radiation measurements, at a university or nuclear physics institute in the country. They have the task of providing field support in cases when the front line officer can not identify an isotope with the RID, or if smuggled nuclear material or source is found, While laboratory type equipment is often available, battery powered, transportable equipment designed for use under field conditions is often missing: A hand-portable HPGe detector with a short cool-down time, coupled to a portable multi channel analyzer, a notebook computer – with long battery life and easy to use expert software for isotope identification and the characterization of Uranium and Plutonium samples under field conditions. In addition a set of portable equipment as used by the front line officers is also provided, since the staff of this group provides equipment training and troubleshooting.
Coordinated Research Project (CRP) is a collection of research agreements and contracts between the IAEA and institutions in Member States, devoted to a specific topic – in this case the improvement of technical means for radiation detection at borders. It is a way for the IAEA to promote R&D in a certain field, to stimulate cooperation and to disseminate information. Recipient countries are awarded research contracts with financial support, about 10-20% of the total project cost (about 10-20 kUSD/year). With industrialized countries, research agreements are concluded. They do not include financial support, except the invitation to the yearly Research Coordination Meetings (RCM). The CRP stretches normally over 3 years. Each contractor provides a yearly progress report and a presentation at the RCM where the work of all contractors is discussed and coordinated. So far two RCM were performed – the first in December 2003 in Vienna and a second in October 2004 in Sochi, Russia.

After the first year, the following major results were obtained [8]:

- Development and implementation of a sensitive, hand-held neutron detector for the localization of weak neutron sources, which had triggered a neutron alarm at a border (Scientific Center Nuclear Physics St. Petersburg); two versions of such a device are now commercially available from two different vendors
- Improvement of Radioisotope Identification Devices - RIDs (Swierk, Poland) – investigation of new scintillator materials to improve the performance of such devices, which do not yet satisfy the requirements of users. Most promising new materials for such applications appear to be LaCl-3, LaBr-3, large volume CZT and LiI(Eu) scintillators
- Demonstration of the use of Radioisotope Identification Devices coupled to a robot to characterize the radiation field of a suspected Radiation Dispersal Device (RDD); research agreement with the German Office for Radiation Protection.
- Demonstration of the use of RIDs for the characterization of radiation sources in legal, border crossing shipments, without opening the shipment container (research contracts with the Kurchatov Institute and the Federal Customs Service of the Russian Federation)
- Development and field test at the Vienna International Airport of a spectral radiation monitor for pedestrians and their luggage. The device identifies automatically medical and other isotopes when passengers pass the exit gate of the arrival area at the airport. This method can reduce considerably the workload of the customs officers in following up a radiation alarm (Atom Institute of the Austrian Universities and Austrian Research Center Seibersdorf)
- In cooperation international experts, equipment users and vendors, the development and harmonization of IAEA Technical Specifications for Border Monitoring Equipment with American Standards [3,9] and national standards of other Member States and international standardization groups has been performed
- Development of documents for Nuclear Forensics Support and for Radiation Monitoring of Mail [10]]
- Development and test of a border monitor (China)
- Identification of nuclear materials in presence of other gamma emitters (Australia, ANSTO)
- Catalogue of gamma spectra for identification software test (JRC Ispra)
- Documents developed under the CRP: Forensics in cooperation with the International Working Group on Illicit Trafficking and UPU and Specifications with experts and user

4. Specific Support Projects

A number of support projects are handled in cooperation with the Department of Technical Cooperation of the IAEA. The funding comes either from the IAEA regular budget or from extra budgetary funds, like the Nuclear Security Fund (NSF). Under such a scheme, support (in the most cases INSServ missions, training and provision of equipment) was provided to various countries in the following areas: Balkan, Newly Independent States (NIS), Middle Asia, Asia, South America and Africa.

In addition, some Member States or Organizations have provided funding to support, through the IAEA, to certain countries of groups of countries with training and equipment. The following examples are given here:

- EU project to support selected Balkan and NIS countries
- Canadian donation to support equipment installation and training at the Chernobyl exclusion zone
- Trilateral support project to support the radiological security of the Olympic Summer games in Greece in 2004 (USA, Greece and IAEA)

5. Summary and Outlook

After 9/11/2001 the International Atomic Energy Agency has, in response to the request of Member States, enhanced its effort to combat illicit trafficking. A wider, more general approach was taken which addresses prevention of and response to any malicious act using nuclear or other radioactive material. The Nuclear Security Plan of Activities was developed and Member States contributed extensively to the Nuclear Security Fund, which was established for the implementation of the Plan. After three years, significant progress on its implementation can be reported.
Presently a follow up plan for the next four years (2006-2009) is being drafted: “Nuclear Security Plan: Protecting Against Terrorism”. It will be a comprehensive effort to help Member States prevent acts of nuclear terrorism, detect attempts to use nuclear and radioactive material for malicious purposes, and develop appropriate responses to these threats. This document will be presented to the IAEA Board of Governors in September for approval.
While in the first three years of the plan, emphasis was given to fact-finding and evaluation, in the next 4 years the focus will be on implementation. This will be done by using a holistic approach, promoting the development of Integrated Nuclear Security Support Plans (INSSP).

6. Acknowledgement

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7. References

[1] N.N.Kravchenko, B.I.Bozhko, *The barrier for 'dirty' materials in journal "Customs" (Official edition of the Federal Customs Service of the Russian Federation), No 7*, 2005.

[2] B.D. Geelhood, J.H. Ely, R.R. Hansen, R.T. Kouzes, J.E. Schweppe, R.A. Warner, *Overview of Portal Monitoring at Border Crossings*, IEEE, 2003

[3] International Atomic Energy Agency, Nuclear Security Series: *Technical / Functional Specifications for Border Radiation Monitoring Equipment*, final draft 2005.

[4] International Atomic Energy Agency, *IAEA-TECDOC-1313*, *“Response to events involving the inadvertent movement or illicit trafficking of radioactive materials”,* IAEA 2002

[5] J.H. Ely, R.T. Kouzes, B.D. Geelhood, J.E. Schweppe, R.A. Warner, *Discrimination of Naturally Occurring Radioactive Material in Plastic Scintillator Material IEEE,* 2004; R.T. Kouzes, J.H. Ely, B.D. Geelhood, R.R. Hansen, E.A. Lepel, J.E. Schweppe, E.R. Siciliano, D.J. Strom, R.A. Warner, *Naturally Occurring Radioactive Materials and Medical Isotopes at Border Crossings*, IEEE, 2004

[6] Boeck H., Schrenk M., *Measurements of radioactive medical isotopes, which could cause innocent alarms at border monitoring systems, Progress report of CRP IAEA*, 2004 and M. Schrenk, R. Arlt, P. Beck, H. Boeck, F. Koenig, T. Leitha, *A Real Time, Isotope Identifying Gamma Spectrometer for Monitoring of Pedestrians*, presented at IEEE NSS-MIC Conference, Rome, Italy, October 2004, submitted to IEEE *Trans. Nucl. Sci.;* David C. Stromswold, Edward R. Siciliano, John E. Schweppe, James H. Ely, Brian D. Milbrath, Richard T. Kouzes, and Bruce D. Geelhood, *Comparison of Plastic and NaI(Tl) Scintillators for Vehicle Portal Monitor Applications, IEEE, 2004;* James H. Ely, Edward R. Siciliano, and Richard T. Kouzes, *Comparison of NaI(Tl) Scintillators and High Purity Germanium for Vehicle Portal Monitor Applications,* IEEE, 2004; D. C. Stromswold, J. W. Darkoch, J. H. Ely, R. R. Hansen, R. T. Kouzes, B. D. Milbrath, R. C. Runkle, W. A. Sliger, J. E. Smart, D. L. Stephens, L. C. Todd, M. L. Woodring, *Field Tests of a NaI(Tl)-Based Vehicle Portal Monitor at Border Crossings,* IEEE, 2004

[7] R. Arlt, A. Kovar, P. Beck, M. Swoboda, K.E. Duftschmid, *Improvement of equipment needed for the detection and characterization of nuclear material and radioactive sources at borders,* IAEA-CN-86/28, 2001 and M. Swoboda, R. Arlt, A. Lupilov, M. Majorov, M. Moszynski, A. Syntfeld
Spectral Gamma Detectors for Hand-Held Radioisotope Identification Devices (RIDs) for Nuclear Security Applications, presented at IEEE NSS-MIC Conference, Rome, Italy, October 2004, submitted to IEEE Trans. Nucl. Sci.

[8] M. Majorov, *Research and development hand-held neutron monitors, Progress report of CRP IAEA*, 2004; Progress Report, IAEA Research Contract No 12596/ Nuclear Security Multi-donors Fund *"Comparative study of new scintillation materials in application to the border monitoring equipment"* The Andrzej Soltan Institute for Nuclear Studies, PL 05-400 Otwock- Świerk, Poland, Marek Moszyński, 15.09.2003 – 14.09.2004; G. Yakovlev, M. Karetnikov, N. Kravchenko, V. Bezverbny, I. Bannykh, V. Bojko, I. Bryagin, A. Dorin, *Feasability of using hand-held isotope identifiers for attribute testing of radioactive sources in shipment containers*, *Progress report of CRP IAEA*, 2004; A. Syntfeld, M. Moszynski, R. Arlt, M. Balcerzyk, m. Kapusta, M. Majorov, P. Schotanus, R. Marcinkowski, M. Swoboda and D. Wolski, *6LiI(Eu) in Neutron and γ-ray Spectrometry-a High Sensitive Thermal Neutron Detector*, presented at IEEE NSS-MIC Conference, Rome, Italy, October 2004, submitted to IEEE Trans. Nucl. Sci.; Zhang, Jizeng Ma, Minchen Han, Yawen Huang, *The vehicle monitor for the detection of radioactive materials*, *Progress report of CRP IAEA*, 2004; Dr. Eng. Akhmad, Yus Rusdian, *Characterization of various survey meters through car-borne survey in Java Island as a basis data for searching orphan sources*, *Progress report of CRP IAEA,* 2004.

[9] N42.32, *American National Standard, Performance Criteria for Alarming Personal Radiation Detectors for Homeland Security*, IEEE, 2004; N42.34, *American National Standard, Performance Criteria for Hand-held Instruments for the Detection and Identification of Radionuclides*, IEEE, 2004; *N42.35, American National Standard Evaluation and Performance of Radiation Detection Portal Monitors for Use in Homeland Security; N42.38, American National Standard, Performance Criteria for Spectroscopy-Based Portal Monitors used for Homeland Security;*

[10] International Atomic Energy Agency, *Guidelines for Monitoring of Radioactive Material in International Mail Transported by Public Postal Operations*, *2005 International Atomic Energy Agency, Nuclear Forensics Support, Reference document for Responding to illicit events involving Nuclear or other radioactive materials (Rev5.)*, 2005.

Session 7

NDA - Monte Carlo Calculations

Calibration of neutron collars for fresh fuel element verification

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Abstract

IAEA inspectors perform all the measurements with neutron collars using the INCC acquisition software. The procedure is based on the application of the cross-reference calibration principle. This principle assumes that all the collars have the same behaviour and their calibration curves differ only on a multiplication factor. Therefore once a calibration curve has been established for a reference collar, this curve can be used for all the collars provided that suitable corrective factors are applied to take into account deviations from the reference conditions due to individual characteristics of the collar (efficiency, source strength) or of the assembly (U loading, burnable poisons, etc.).

The scope of the project described in this paper is to use Monte Carlo calculations to analyse and improve the calibration of neutron collars used for the verification of fresh fuel elements.

Some of the possible aspects studied are listed below:

- *extension of the existing calibration curves to higher enrichments (up to 5%)*
- *burnable poison correction for fuels with Gd poisoned rods*
- *differences between collar models (UNCL and UNCL-II)*
- *effects due to different fuel geometry (smaller PWR elements, WWER)*

Keywords: neutron counting, calibration, NDA, Monte Carlo simulation.

Theme 11: Measurement techniques and standards

Topic: NDA – Monte Carlo Calculations or NDA – Quantitative measurements

1 Introduction

The present work has been performed under the EC support program to IAEA, task EC-A-1505.

The scope of this activity is to use Monte Carlo calculations to analyse and improve the calibration of neutron collars used for the verification of fresh fuel elements. Some of the possible aspects to study are listed below:

- extension of the existing calibration curves to higher enrichments (up to 5%)
- burnable poison correction for fuels with Gd poisoned rods
- differences between collar models (UNCL and UNCL-II)
- effects due to different fuel geometry (smaller PWR elements, WWER)

IAEA inspectors perform all the measurements with neutron collars using the INCC acquisition software [1]. The procedure is based on the application of the cross-reference calibration principle [2]. This principle assumes that all the collars have the same behaviour and their calibration curves differ only on a multiplication factor. Therefore once a calibration curve has been established for a reference collar, this curve can be used for all the collars provided that suitable corrective factors are applied to take into account deviations from the reference conditions due to individual characteristics of the collar (efficiency, source strength) or of the assembly (U loading, burnable poisons…).

In order to reduce the impact on the inspection procedures, it is requested that the improvements produced by this analysis:

- do not affect the application of the cross-reference calibration principle
- requires minimum, or possibly none, modifications on the INCC software

This last condition means that the proposed modifications should not alter the formalism, but when possible the improvement should require just the modification of parameters entered by the user.

2 PWR fuel elements in thermal mode

2.1 Extrapolation of the calibration curve to higher enrichments

The reference collar was calibrated with the LANL reference fuel assembly having ^{235}U enrichment of 3.2% corresponding to a linear mass of ~39 g/cm. Lower enrichments were obtained by progressive replacements of enriched uranium pins with depleted uranium pins, down to a linear mass of 10 g/cm. So the reference calibration curve was established in the range 10 to 39 g/cm.

Nowadays the fuel elements have larger uranium loading with enrichments above 4%. Typical linear mass of modern fuels can easily extend up to 55 g/cm or more. It should be verified whether the extrapolation of the reference calibration curve well above its validity range is acceptable or a new curve is needed.

Not having the possibility to test this experimentally, we have performed Monte Carlo simulations using the MCNP-PTA code developed at JRC [3].

As a first step, we wanted to reproduce the experimental data of the reference calibration exercise. The main difficulty in this step was the unknown value of the MRC-95 AmLi source intensity. In the LANL calibration report [2] a value of 39600 is given. In a more recent report [4] a much lower value is reported (33500). This latter estimation is confirmed by cross-reference of Euratom sources measured at LANL and in PERLA that hinted to a value of 33850 [5].

Lacking of a fully trusted value of the reference source, we decided to proceed on the opposite way. We fixed the source intensity to the value that reproduced the correct experimental rate with the reference experimental conditions (35000 s^{-1}). In this way all our calculations will be automatically “cross-referenced” to the LANL reference measurements.

Then a computational calibration has been established under the following conditions:

- UNCL-II collar (CANBERRA model JCC-73) in thermal mode (no Cd)
- AmLi source with intensity of 35000 s^{-1}
- Typical 17x17 PWR fuel assembly with constant uranium loading of 1261.5 g/cm (corresponding to an uranium mass corrective factor of 0.982 with respect to the LANL assembly)
- Variable enrichment from 1% to 5% with 0.5% steps covering the linear mass range from 13 to 63 g/cm

The results are shown in figure 1. The LANL reference calibration curve is reported in black, the markers show the computed values and the pink line is the best-fit of the computed values with the usual hyperbolic function

$$R = a * m / (1 + b * m)$$

obtained from the Deming program. The a and b parameters of the two calibration curves are compared in table 1.

We see that the two curves overlap in the range 30 to 45 g/cm, where they agree within 1%, corresponding to the classical range of ~3% enriched fuel assemblies. At higher enrichments the two curves start to diverge slightly. In the region around 55 g/cm (corresponding to nearly 4.5% enrichment) there is a difference of 2% in Reals, which amplifies to a 5.5% difference in mass due to the slope of the curve.

Figure 1 – Comparison between experimental and computational calibrations for PWR fuels in thermal mode

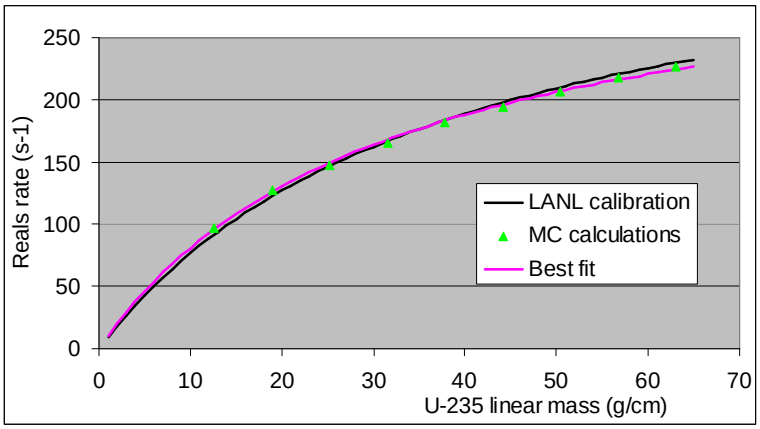


Table 1 – Calibration parameters for UNCL-II collars (PWR in thermal mode)

Calibration parameters	Experimental (LANL)	Computational (JRC)
a	9.646	10.6
b	0.0261	0.0314
var (a)	4.66E-02	3.96E-02
var (b)	1.69E-06	1.11E-06
covar (a:b)	2.78E-04	2.06E-04

2.2 Burnable poison correction factor

When the assembly contains rods poisoned with an absorber (such as gadolinium) the measured Reals rate must be corrected to take into account the neutrons absorbed by the poison. The k3 correction factor is a function of:

- the total number of fuel pins in the measured assembly (and in the reference assembly)
- the number of poisoned pins
- the poison material and content
- the average uranium enrichment in the assembly

The correction law used by the INCC software is:

$$k_3 = 1 + n * (N_{ref}/N) * a * [1 – exp\text{-}(P*Gd)] * (b – c * Enr)$$

where:

- N_{ref} and N are the number of fuel pins in the reference and actual assemblies
- n is the number of poisoned pins
- Gd is the gadolinium loading (in weight percent) in the poisoned pins
- Enr is the average fuel enrichment
- P is the poison absorption factor (constant depending on the type of absorber)
- a, b and c are free parameters to be provided by the user

The LANL report [2] provides a set of parameters to be used in the above equation for the calculation of the correction factor. These were obtained by fitting experimental data obtained with the available poisoned rods (12 rods with 5.2% Gd weight) and available enrichments (<3%).

Today inspectors have to measure fuel assemblies with higher enrichments and much higher poison loading (up to 24 rods and to 10% Gd content). It is therefore logical to expect that the LANL correlation needs to be updated to cope with the new fuel conditions.

In a previous work [5], JRC had already performed a preliminary analysis with Monte Carlo calculations of the burnable poison correction. In that case a grid of 20 poison loading configurations (4-8-12-16-20 rods with 2-4-6-8% Gd) was computed with a 3% enriched fuel. The values were fitted to get the P and a parameters. This resulting law was found in fairly good agreement with the LANL correlation for 3%-enriched fuel. Then for a fixed selected configuration (8 rods with 8% Gd) the enrichment was changed and the results were fit to get the b and c parameters. We found large discrepancies on these parameters with the LANL values. Therefore the Monte Carlo calculations seem to suggest deviations from the LANL correlation when the enrichment increases significantly above 3%.

In this work we decided to repeat the analysis in a more systematic way, including the extreme cases that could be found in reality. Then we computed a new calculation grid with 90 different configurations deriving from all the possible combinations of:

- three different enrichments (3-4-5%)
- six lattice configurations (4-8-12-16-20-24 poisoned rods)
- five poison loadings (2-4-6-8-10% Gd weight)

Then the entire set was fit with the above equation in order to get simultaneously the values of the four parameters. Theoretically the poison absorption factor (P) should be a constant, but we preferred to leave it as a variable in order to get an optimal fit of the data. The final values of the parameters are shown in Table 2. Extensive detailed results are collected in a JRC report [6].

Table 2 – Correlation law for burnable poison correction (PWR in thermal mode)

Fitting parameters	Experimental (LANL)	Computational (JRC)
Nref	204	264
P	0.647	0.835
a	0.0213	0.0219
b	2.27	1.24
c	0.40	0.13

Comparing the correction factors obtained from Monte Carlo calculations with the values computed using the LANL correlation, we see that both of them give consistent results with 3%-enriched fuel. This confirms the quality of the Monte Carlo simulation. In fact the LANL correlation was mainly obtained with experimental measurements done with 3.2%-enriched fuel and therefore it has to be considered reliable and trusted under these conditions.

For higher enrichments there are increasing deviations between the Monte Carlo calculations and the LANL correlation. Typical deviations are of the order of 2% for 3% enrichment, 5% for 4% enrichment and 10% for 5% enrichment. The JRC correlation fits the computed data with a typical error of 1% in the entire set of configurations.

3 PWR fuel elements in fast mode

3.1 Extrapolation of the calibration curve to higher enrichments

The calculation of the calibration curve for measurements of PWR fuel elements in fast mode (with Cd liners) has been performed in the same way described in the previous chapter for the thermal mode. The results are shown in figure 2. The a and b parameters of the two calibration curves are compared in table 3.

We see that the two curves overlap in the range 30 to 50 g/cm, where they agree within 1%, corresponding to the classical range of ~3% enriched fuel assemblies. At higher enrichments the two curves start to diverge slightly, but the difference exceed 2% only above 60 g/cm (corresponding to enrichments above 4.7%). Basically the two curves agree in most of the practical range, so we do not expect dramatic differences in the results. The calculations substantially confirm the validity of LANL calibration even above the calibration range.

Figure 2 – Comparison between experimental and computational calibrations for PWR fuels in fast mode

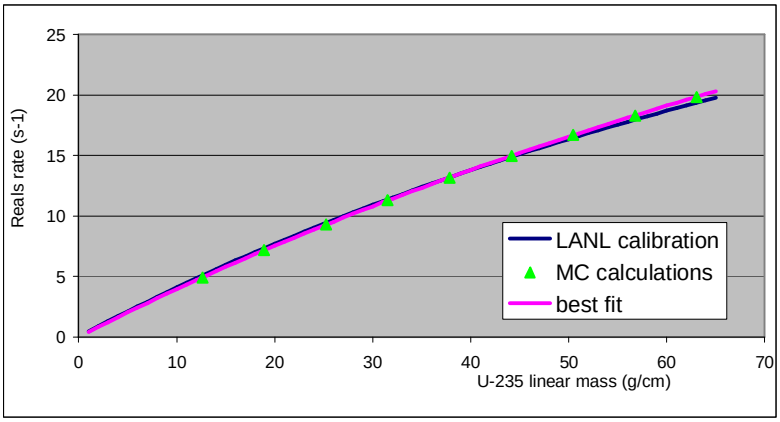


Table 3 – Calibration parameters for UNCL-II collars (PWR in fast mode)

Calibration parameters	Experimental (LANL)	Computational (JRC)
a	0.4373	0.415
b	0.00671	0.00504
var (a)	2.28E-05	1.40E-04
var (b)	2.29E-07	5.96E-07
covar (a:b)	2.18E-06	8.87E-06

3.2 Burnable poison correction factor

Differently from the thermal mode, INCC assumes that the poison correction factor in fast mode does not depend from the enrichment. Therefore the correction law reduces to the simplified form:

$$k_3 = 1 + n * (Nref/N) * a * [1 – exp-(P*Gd)]$$

This is equivalent to impose b=1 and c=0 in the general equation.

According to the previous statement, the calculation of burnable poison correction factors could be done at whatever enrichment, but in order to verify the correctness of the assumption we repeated the calculation for two different enrichments. We used in total 60 configurations with the following characteristics:

- two different enrichments (3 and 5%)
- six lattice configurations (4-8-12-16-20-24 poisoned rods)
- five poison loadings (2-4-6-8-10% Gd weight)

Then the entire set was fit with the above equation in order to get simultaneously the values of the two parameters (P and a) and results are in table 4. We tested also a fit with all four free parameters, but the results are quite similar, confirming so the initial assumption that the correction factors do not depend from the enrichment.

Notwithstanding the parameters seem completely different, the factors computed with the two laws agree quite well among them and with the calculations. The new law fits the computed data typically within 1%. LANL correlation starts to deviate slightly only for pins with high Gd loading (8-10%) and gives an overall typical deviation of 2-2.5%.

As a conclusion, we can state that the new calibration computed with Monte Carlo could possibly improve slightly the evaluation of PWR measurements in fast mode, but not significantly. LANL calibration seems to be adequate to this kind of measurements and there is no clear evidence of need to replace it.

Table 4 – Correlation law for burnable poison correction (PWR in fast mode)

Fitting parameters	Experimental (LANL)	Computational (JRC)
Nref	204	264
P	0.647	0.034
a	0.00295	0.0177
b	1.0	1.0
c	0.0	0.0

4 BWR fuel elements in thermal mode

4.1 Extrapolation of the calibration curve to higher enrichments

For the measurement of BWR fuel elements we have modelled the following conditions:

- UNCL collar (CANBERRA model JCC-71) in the BWR configuration that means with the side slabs fixed in the positions generating the narrow cavity (16.5 cm)
- AmLi source with intensity of 35000 s^{-1}
- Typical 9x9 BWR fuel assembly with constant uranium loading of 498.6 g/cm (corresponding to an uranium mass corrective factor of 0.967 with respect to the LANL assembly)
- Variable enrichment from 1% to 5% with 0.5% steps covering the linear mass range from 5 to 25 g/cm

The results are shown in figure 3. The a and b parameters of the two calibration curves are compared in table 5.

The two curves overlap in the range 12 to 18 g/cm, where they agree within 1%, corresponding to the classical range of ~3% enriched fuel assemblies. At higher enrichments the two curves start to diverge slightly, but the difference exceed 2% only above 22.5 g/cm (corresponding to enrichments above 4.5%).

.Fig. 3 – Comparison between experimental and computational calibrations for BWR fuels in thermal mode

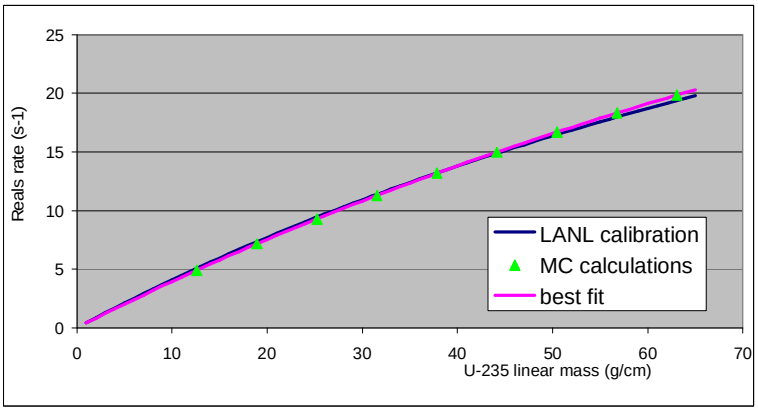


Table 5 – Calibration parameters for UNCL collars (BWR no Cd)

Calibration parameters	Experimental (LANL)	Computational (JRC)
a	24.91	27.49
b	0.0680	0.0822
var (a)	1.09E-01	6.90E-02
var (b)	4.45E-06	2.39E-06
covar (a:b)	6.85E-04	3.95E-04

4.2 Burnable poison correction factor

For BWR elements, INCC uses the same formalism used for PWR elements. LANL recommends different values for the parameters appearing in the correlation law (with the exception of the poison absorption factor, P, which is kept constant and equal to 0.647).

For BWR measurements in thermal mode, we have computed the correction factors by fitting a grid of 48 simulated cases corresponding to the possible combinations of:

- three different enrichments (3-4-5%)
- four lattice configurations (4-8-12-16 poisoned rods)
- four poison loadings (2-4-6-8% Gd weight)

The final values of the parameters are shown in Table 6. LANL values were obtained from measurements performed with 3.2% enriched fuel and up to 12 rods poisoned with 5.2% gadolinium. The computed values agree very well with the LANL values in this range. Differences appear with higher gadolinium loading and especially at higher fuel enrichment. This is somehow consistent with the results of PWR fuel elements in thermal mode: the main discrepancy between our computed values and LANL correlation derives from the delta factor (the enrichment correction factor).

The new proposed correlation law reproduces the computed values with reasonable accuracy (a standard deviation of 5%). The fit is somehow poorer with respect to the PWR case. This is due to a slight non-linearity of the correction factor as a function of the number of poisoned rods, probably due to the larger ratio of poisoned/unpoisoned rods and to the shorter distance between poisoned rods causing interference (shadow) effects.

Table 6 – Correlation law for burnable poison correction (BWR-thermal)

Fitting parameters	Experimental (LANL)	Computational (JRC)
Nref	76	76
P	0.647	1.04
a	0.0572	0.0577
b	1.92	1.48
c	0.29	0.13

5 BWR fuel elements in fast mode

5.1 Extrapolation of the calibration curve to higher enrichments

To complete the study, we analysed the case of BWR elements in fast mode. The same cases of the previous chapter have been run with the addition of cadmium liners. Figure 4 and table 7 show the results. Here the two curves start to deviate already for linear masses above 17 (corresponding to an enrichment of 3.5%) and the difference grows up to 4-5% for enrichments beyond 4.5%.

Figure 4 – Comparison between experimental and computational calibrations for BWR fuels in fast mode

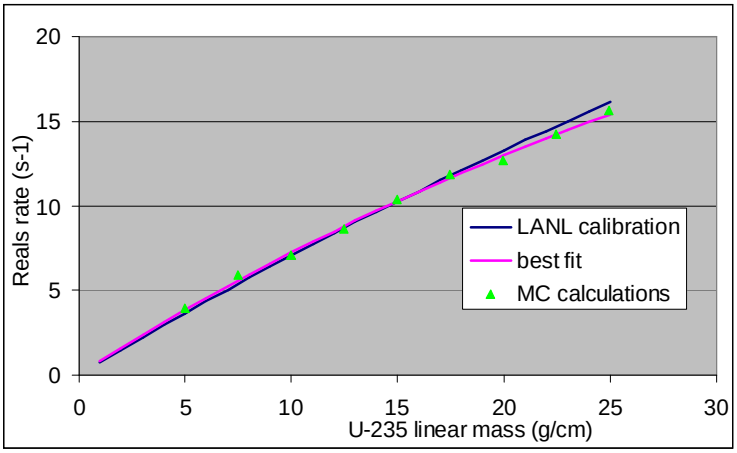


Table 7 – Calibration parameters for UNCL collars (BWR with Cd)

Calibration parameters	Experimental (LANL)	Computational (JRC)
a	0.7582	0.819
b	0.00705	0.01318
var (a)	1.12E-04	6.01E-04
var (b)	1.57E-06	5.00E-06
covar (a:b)	1.31E-05	5.25E-05

5.2 Burnable poison correction factor

As already said before, it is generally assumed that the poison correction factors do not depend from the fuel enrichment when measuring in fast mode. We have already verified the correctness of this statement for PWR elements, so we assumed it valid also for BWR fuel.

Therefore we have computed the corrective factors only with an enrichment of 4% for the 16 cases deriving from four lattice configurations (4-8-12-16 poisoned rods) with four gadolinium loading (2-4-6-8%). The results are summarised in table 8.

As in the case of PWR fast mode, despite a large apparent difference between the parameters, the factors computed with the two laws agree quite well among them and with the calculations. The new law fits the computed data typically within 1% whereas the LANL correlation agrees typically within 2%.

Table 8 – Correlation law for burnable poison correction (BWR in fast mode)

Fitting parameters	Experimental (LANL)	Computational (JRC)
Nref	76	76
P	0.647	0.062
a	0.00661	0.0123
b	1.0	1.0
c	0.0	0.0

6 Effects of detector/assembly geometry

6.1 Comparison UNCL and UNCL-II

IAEA inspectors use two different models of neutron collar for LWR assemblies: the modular UNCL (CANBERRA model JCC-71) with variable geometry and the fixed geometry UNCL-II (CANBERRA model JCC-72 for BWR and JCC-73 for PWR).

The cross-reference method assumes that the same calibration curve applies to both the models. The experimental data reported in reference [2] show that the measurements done with the collars LANL-1 (UNCL) and LANL-3 (UNCL-II) lie on the same curve provided that the LANL-1 data are multiplied for an appropriate normalisation constant.

To confirm the assumption, we repeated for a UNCL collar the same calculations described in section 2.1 to establish the UNCL-II calibration. In the entire mass range, the ratio of Reals between the two collar models remained constant within the statistical error (1.463±0.014). This confirms that the same calibration curve can be applied to both models. The k_2 factor will take care of the different detection efficiency of the two models.

6.2 Positioning of BWR fuel inside the collar cavity

The JCC-72 collar for BWR fuel, the JCC-73 collar for PWR fuel and the modular JCC-71 in PWR configuration have all a square cavity nearly 2 cm larger than the fuel assembly. So they do not present any problem to the correct positioning of the fuel element at the centre of the cavity.

On the contrary the JCC-71 in BWR configuration has a rectangular cavity: 165 x 235 mm. Therefore the typical square BWR element with a side dimension of 13 cm can be displaced inside the cavity over a longitudinal length of 10 cm.

The response of the collar is strongly dependent from the position of the element inside the cavity. The maximum count rates are obtained when the fuel is as close as possible to the AmLi source, because the probability of inducing fission is maximised. Positioning the element at the centre of the cavity or at the opposite extreme will cause a decrease of the Reals rate of 6% and 15% respectively. Therefore it is of paramount importance that the assembly is located in a defined (and reproducible) position. The position recommended in the LANL report is the one leaving a 1-cm gap between the assembly and the collar slab hosting the source. In this position the experimental calibration curves were obtained. All the Monte Carlo calculations described above have been also done in this position.

For a correct interpretation of experimental results it is fundamental that the inspectors follow this recommendation on the assembly position.

6.3 PWR assembly size

Fuel elements used in PWR reactors have generally a square geometry, but they may have different lattice arrangements typically ranging from 14x14 to 18x18 rods.

Nevertheless in most cases the overall dimension of assemblies are generally very similar (nearly 21.5 cm), the rod pitch decreasing when increasing the number of rods: for instance 1.26 cm in a 17x17 lattice or 1.43 in a 15x15. This results just in a slightly different distribution of fissile inside the same "effective" volume. The effect of this geometrical redistribution of fuel in the counter cavity is considered negligible and the same calibration curve is applied to all the fuel element types.

First of all we wanted to verify this assumption, comparing the calculations of two assemblies having the same mass loading in a 15x15 versus a 17x17 arrangement. The Reals rates in the two calculations differed less than 0.5%. Since this is roughly the order of the precision (statistical error) of our calculations we cannot exactly provide a precise value for this geometrical effect, but we can anyway state that it is very small and certainly lower than 1%. Therefore the assumption is justified.

Unfortunately there are cases where the condition of constant assembly dimension is not satisfied. For instance there is a model of a 14x14 assembly having the same pitch of the standard 15x15, resulting in a smaller total assembly width (20.2 cm instead of 21.5). In this case the fuel is not redistributed in the same volume, but concentrated in a smaller effective volume. This causes several consequences:

- the most important is the reduction of the solid angle under which the source sees the element, reducing so the probability of inducing fission
- fission occurs in a reduced region of the cavity, altering so the average detection efficiency
- leakage and self-multiplication are also slightly modified
- from a practical point of view the larger gap between assembly and collar makes more difficult the correct positioning of the assembly, introducing a positioning error and a consequent lower reproducibility of the measurement

A calculation was done for a narrow 14x14 element positioned at the centre of the cavity of a JCC-73 collar interrogated with the MRC-95 source. The U-235 linear mass in the calculation was 42.09 g/cm. The computed Reals rate resulted to be 172.98 cps. The expected Reals rate in a standard assembly having the same fissile loading is 193.47 cps. The reduction in count rate due to the geometry change is as high as 12%.

We have also calculated the positioning error deriving from a 1-cm displacement of the small assembly inside the cavity with respect to the central position. The largest deviation results when the assembly is displaced towards the source causing an increase of 2% in the Reals.

We see clearly the need of using a corrective factor accounting for the assembly width change. The corrective factor has to be computed case by case for each individual assembly type.

6.4 Hexagonal WWER assemblies

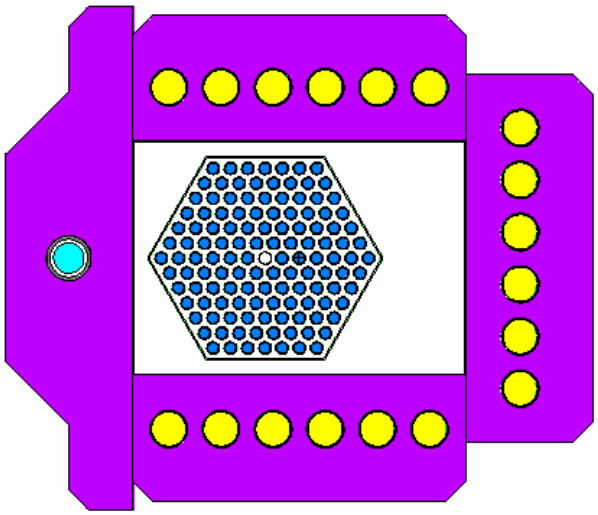
The fuel elements of standard western LWR reactors are typically composed by bundles of fuel rods disposed in a square lattice. The Russian-type WWER reactors (both the older WWER-440 and the more recent WWER-1000) are loaded with hexagonal bundles composed by fuel rods disposed in a triangular lattice. Table 9 reports the main characteristics of WWER fuel elements.

Comparing the assembly dimensions with the collar cavity it is clear that WWER-1000 elements will not fit into any of the existing commercial collars. Special dedicated collars with larger cavity have to be developed for the larger assembly model. On the contrary the WWER-440 fits very well in the cavity of the UNCL collar in BWR configuration. Of course it would also enter into the PWR configurations of both the collar models, but in this case the measurement geometry will be less favourable. So we will assume the UNCL collar in BWR configuration as the optimal measurement condition for WWER-440. Figures 5 shows the recommended assembly disposition inside the cavity, with the wrapper corner placed at 1 cm from the collar slab hosting the AmLi source. This measurement configuration has been used to build the mathematical model for all the Monte Carlo simulations performed in this study.

Table 9 – Characteristics of WWER fuel elements

	WWER-440	WWER-1000
Fuel bundle characteristics		
Assembly width (between sides) (cm)	14.40	23.80
Assembly length between corners (cm)	16.63	27.48
Wrapper material	Zr+1%Nb	Zr+1%Nb
Hexagonal lattice parameters		
Number of rod rings per element	7	11
Number of guide/hollow tubes	1	14
Number of fuel rods	126	317
Rod pitch (cm)	1.220	1.276
Fuel rods characteristics		
Fuel density (g/cm^3)	10.6	
Pellet diameter (cm)	0.753	
Inner pellet hole diameter (cm)	0.140	
Internal cladding diameter (cm)	0.845	
External cladding diameter (cm)	0.910	
Active length (cm)	118.4	
Cladding material	Zr+1%Nb	

Figure 1 – Optimal measurement configuration for a WWER-440 element in a UNCL collar



A series of calculations have been performed under the following assumptions:

- UNCL collar (CANBERRA model JCC-71) in BWR configuration both in thermal and fast mode
- AmLi source with intensity of 35000 s^{-1}
- WWER-440 fuel assembly with constant uranium loading of 534.08 g/cm (corresponding to an uranium mass corrective factor of 0.941 with respect to the LANL reference assembly)
- Variable enrichment from 1% to 5% with 0.5% steps covering the linear mass range from 5.3 to 26.7 g/cm

Complete detailed results are reported in [7].

According to the results of the Monte Carlo calculations, we can suggest two possible ways to calibrate a collar for the measurement of WWER-440 fuel elements.

a) Use of a correction factor for assembly geometry

This option derives from the quasi-parallel behaviour of the BRW and WWER response functions.

When a collar has been already cross-calibrated with respect to the reference LANL collar, it is convenient to make use of all its known parameters. Therefore the calibration curve of the BWR fuel element can be used provided that a suitable correction factor k_5 is introduced accounting for the different assembly geometry.

Notwithstanding the ratio of the response functions is slightly dependent from the fissile linear mass, its variation in the most frequent mass range is quite small. Therefore it is possible to assume constant correction factors equal to:

- $k_5 = 1.06 \pm 0.01$ for thermal mode
- $k_5 = 0.94 \pm 0.01$ for fast mode

b) Specific computational calibration curve

In the case the collar has not been cross-calibrated with respect to the LANL reference collar, and the k_2 factor is unknown, it is more convenient to use an absolute calibration curve as derived from the Monte Carlo calculations.

In this case the k_2 factor has to be set to 1.000, introducing the assumption that all the collars of the same model have the same efficiency. Experience teaches that this is not true in reality. Analysing the experimental data obtained measuring the same reference assembly with the same source and many similar collars of the same model, we can see fluctuations of the response functions of individual collars up to 5%. This means a clear limitation on the final accuracy of the method, but has the advantage of being applicable to any non-calibrated collars.

The calibration parameters have been obtained by a least-square best fit of the values computed with MC. The main difference to be accounted for, between the computational model and the real measurement device, is the AmLi source strength, described by the k_0 factor. Calculations have been done for a source intensity of 35000 s^{-1} corresponding to the intensity that reproduces correctly the results of the MRC-95 LANL reference source. In order to compute the appropriate k_0 the user should know:

- either the absolute intensity of the source in neutrons per second
- or to know the relative intensity of the source with respect to the MRC-95

The calibration parameters for the measurement of WWER-440 fuel assemblies with UNCL collars in BWR configuration are reported in table 10.

Table 10 – Calibration of UNCL collar for WWER-440 elements

Calibration parameters	Thermal mode	Fast mode
a	15.15	0.441
b	0.078	0.012
var (a)	1.34e-2	4.0e-5
var (b)	1.16e-6	1.4e-6
covar (a,b)	1.22e-4	6.6e-6

7 Conclusions

From the global analysis of the results covering all the measurement modes we can derive the following conclusions:

- there is no evidence of dramatic discrepancies when extrapolating the LANL calibration curves to higher enrichments above the calibration range for non-poisoned fuel elements

- according to Monte Carlo calculations, the LANL calibration generally tends to overestimate the Reals rate, and therefore underestimate mass, of few percents in all the configurations, with the exception of the PWR fast mode, when the enrichment is above 4%
- new correlation laws, derived from MC calculations, could bring some (modest) improvements in the measurement of non-poisoned fuels with enrichments beyond 4%
- we have detected a non-negligible effect due to the fuel geometry (i.e. the assembly width) that was not considered up to now and must be accounted for
- the results underline significant inconsistencies between the poison corrective factors computed with Monte Carlo and those obtained with LANL correlation, mostly deriving from the dependence of the correction from the fuel enrichment (for the thermal modes only)
- the LANL correction factors for fast modes are accurate and the assumption of independence from the fuel enrichment is valid
- new correlation laws have been produced for use with INCC that are expected to impact significantly the results of measurements of poisoned fuel elements in thermal mode with enrichments >3.5%
- these laws have been obtained for an axially homogeneous poison distribution and must be used carefully when dealing with axially heterogeneous assemblies
- two possible methods have been drawn to extend the method to hexagonal WWER-440 fuel elements measured with UNCL collar in BWR mode

8 References

[1] B. Harker, M. Krick: "*INCC Software User Manual*", LA-UR-99-1291, (1998).

[2] H.O. Menlove et al.: "*Neutron Collar Calibration and Evaluation for Assay of LWR Fuel Assemblies Containing Burnable Neutron Absorbers*", LA-11965-MS, (1990).

[3] P. Peerani, M. Looman: "*Computational modelling of NDA instruments for nuclear safeguards*", Proceedings of the 24th ESARDA Annual Meeting, Luxembourg, 28-30 May 2002.

[4] P. Rinard, H. Menlove: "*Calibration of the IAEA AWCC for Measurements of the Bruce-A Unirradiated Booster Rods*", LA-UR-98-4362, (1998).

[5] M. Looman et al.: "*The Effect of Gadolinium Poison Rods on the Active Neutron Measurement of LEU Fuel Assemblies*", Proceedings of the 23rd ESARDA Conference, Bruges (B), 8-10 May 2001.

[6] P. Peerani: "*Computational calibration of UNCL neutron collars for LWR fresh fuel elements*", JRC Technical Note I.04.142, October 2004.

[7] P. Peerani: "*Calibration of UNCL neutron collars for WWER-440 fresh fuel elements*", JRC Technical Note I.05.07, January 2005.

Scrap Neutron Multiplicity Counter: Design and Manufacture

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Abstract:

*Mixed (plutonium and uranium) oxide is processed in many of the facilities of the nuclear fuel cycle. Consequently the fissile mass of a large number of MOX and **MOX scrap** samples need to be accounted for and verified periodically in each facility. Neutron counters are used both by plant operators and safeguards inspectors for this purpose. The methods commonly used are neutron coincidence counting (pair correlation technique) or neutron multiplicity counting (triple correlation technique) [1]. For MOX scrap samples the preferred method is often neutron multiplicity counting because three quantities are measured (singlet, doublet, and triplet rates). This allows for the determination of two parameters in addition to the fissile mass. This is useful in cases where the alpha ratio is unknown (for example due to chemical treatments), and in cases where the counter efficiency cannot be anticipated (for example due to a hydrogenous sample matrix).*

In the design of this new counter the primary objective was to achieve a maximum efficiency. However, other factors must be taken into consideration such as cost, size, weight, and in particular the requirements of the analytical model [1]. One important objective is to reduce the detector die-away time thus decreasing the number of uncorrelated (accidental) signal multiplets and consequently improving the "signal-to-noise" ratio.

Keywords: multiplicity counting; MOX scrap, high efficiency neutron counter.

1. Introduction

The counter's parameters to be defined were:

- The assay chamber size.
- Helium-3 gas pressure and composition.
- Number and distribution of the ^{3}He tubes.
- Material and extension of each zone.
- External dimensions of the counter.

An iterative process for the optimisation of the neutron detector has been applied due to the inter-connection among these parameters. It means that when one of these points is modified a re-adjustment of the rest of them is required.

The physical size of the sample container determines the **assay chamber size**. Moreover, the taller is the sample container and the longer become the required **^{3}He tubes.** The use of long ^{3}He tubes helps to keep a **flat vertical efficiency profile** throughout the assay chamber. This is important because the container fill height is usually variable from sample to sample. For containers with distributed Pu sources, a flat spatial efficiency is even more important to reduce deviations from the point model that affect the multiplicity analysis.

For scrap and waste materials with low Pu content, the **detection limit** is an important criterion. It is a function of the detector efficiency, background count rate, and counting time. An **external shield** may be required to reach a low detection limit. To achieve for typical samples a specific assay relative standard deviation in reasonable counting times of 15 to 30 min, a **high detection efficiency** is required, which implies **a large number of ^{3}He tubes.**

For high Pu mass samples other counter properties become important. The **counter die-away time** should be low to minimize the accidental coincidence background, and the electronics should have **low counting dead time** to minimize counting losses. The number of ^{3}He tubes per amplifier should be small.

A Cd-lined sample well is used to forbid the return of thermal neutrons from the polyethylene in order to reduce the **self-multiplication** in large samples. This also improves the **criticality safety** of the counter.

Other **matrix effects** present in small or large samples include (α,n) reactions and neutron moderators or poisons. For high (α,n) reaction rates, low counter die-away time helps to reduce the accidental coincidence background. **Flat energy spectrum efficiency** is also important so that the counter has roughly the same efficiency for detecting neutrons from spontaneous fission and from (α,n) reactions. If the sample matrix contains water or other hydrogenous materials that moderate neutrons and reduce their energy spectrum, the counter should definitely have a flat energy response efficiency to mitigate the matrix effects. The effect of neutron poisons in the sample is not easily observed, but a flat energy spectrum efficiency will reduce the effect of a shift in the emitted neutron energy spectrum, and a Cd-lined sample well will dampen the effect of thermal neutron capture in the poisons.

Samples with high Am content are not only strong sources of (α,n) neutrons, but also strong sources of 60-keV gamma rays. To avoid exceeding the ^{3}He tube dose limit of about 1 R/h, it is sometimes helpful to use a thick cadmium liner on the sample well to increase the attenuation of these gamma rays.

During the first year of the project Monte Carlo calculations were performed to optimise the counter design: materials, dimensions, and specially the quantity, the gas composition and the geometrical distribution of the proportional counters inside the detector. The goal was to respect the different points influencing the parameters of the mathematical model of Hage [1]. During the second year many experimental tests have been made with the aim of validating the results obtained by Monte Carlo calculations.

Once the theoretical study was finished and the final dimensions and positions of the different layers of materials used were established, the fabrication part could start. Practically the complete mechanization process was carried out at the Central Workshop of the Joint Research Centre (JRC - Ispra (Italy)). The constructive designs of the different mechanical parts were also studied and developed inside the JRC structure by the Traceability and Vulnerability Assessment (TRVA) Unit of the Institute for the Protection and Security of the Citizen (IPSC) using information extracted from Monte Carlo calculations.

The dimensions of the SNMC are 1447 mm height x 961 mm diameter, and it weights approximately 1200 kg. A total of 125 proportional counters are allocated inside, giving a total neutron detection efficiency of around 64%.

Figure 1 gives you an idea about the different components of the SNMC before being assembled by the Central Workshop team. Figure 2 illustrates one of the steps of the assembling process, in particular the insertion of the pre-moderation zone inside the moderation sector. This picture also shows the multilayer structure of the SNMC

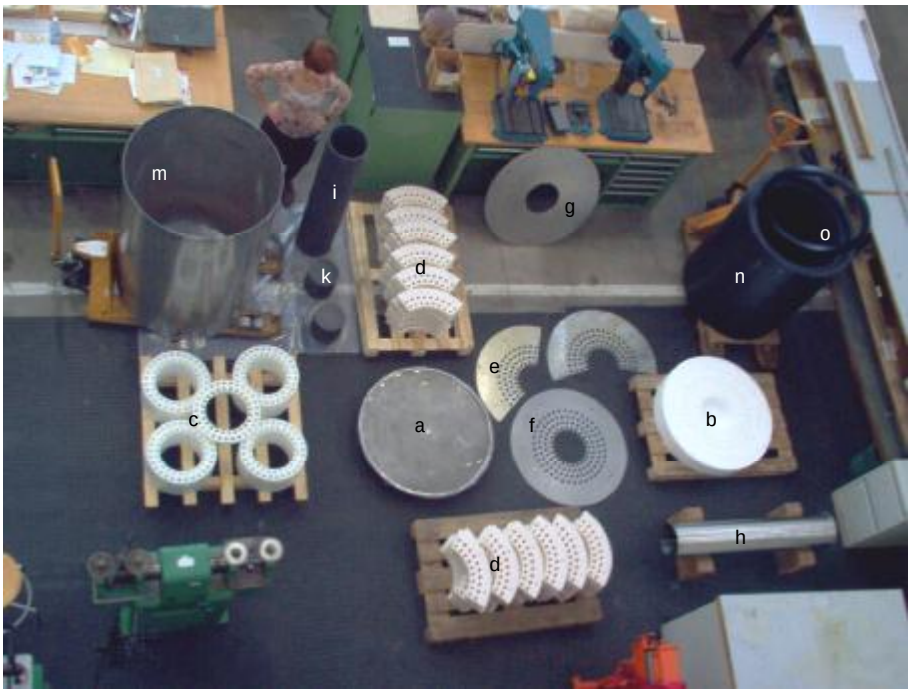

Figure 1 – General view of all the peaces forming the SNMC; **a**- stainless steel base; **b**- bottom piece of polyethylene; **c**- moderation zone in polyethylene; **d**- post-moderation zone in Kynar; **e**- horizontal layer of cadmium; **f**- lower part of the junction box in aluminium; **g**- upper layer of the junction box in aluminium; **h**- part of the pre-moderation zone: stainless steel cylinder; **i**- part of the pre-moderation zone: graphite cylinder; **k**- Graphite cylinders forming plugs before being lined; **m**- external stainless steel cylinder; **n**- part of the external shield in black polyethylene; **o**- reflector made in black polyethylene.

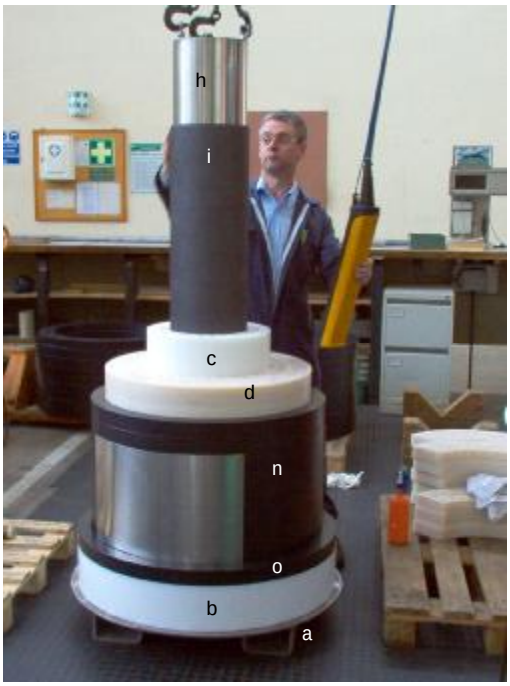

Figure 2 – View of one of the steps of the assembling process corresponding to the incorporation of the pre-moderation zone (the different layers of materials are described in figure 1).

2. Number of ^{3}He detectors.

It is known that the final neutron detection efficiency is proportional to the number of ^{3}He detectors used. After the analysis of several calculations, it has been observed that the optimal distance between the centres of two ^{3}He detectors is around 4 cm. In fact, this corresponds to a thickness of only ≈1.5 cm of effective moderator between two detectors. When detectors are much more closer than those 4 cm, detector shadowing enters into the play and the system's efficiency is reduced. Furthermore, in such conditions the quantity of moderator is insufficient to allow neutrons to reach thermal energies. On the other hand, if the distance between detectors is selected much larger than 4 cm, the moderator will absorb a considerable quantity of neutrons.

The thermal-neutron transport mean free path (mfp) has been calculated using as diffusion factor in polyethylene of D_o=2600 cm^2s^{-1} [2] and the result obtained was 3.2 cm. Otherwise, the scattering mfp is one of the parameters calculated by MCNP and printed in table 126 of the output file. Obviously this result is strongly depending on the energetic neutron spectrum and increases when neutrons flow deeper inside the detector. Typically the quantity of moderator necessary to guarantee the neutron thermalisation without having a large contribution of the neutron absorption factor should be bigger than the scattering mfp but smaller than the absorption one.

Initially and due to geometrical reasons a solution with 32 tubes per ring was studied. A set of MCNP calculations putting a different number of detectors in each ring in order to improve the total efficiency and, respecting only the fact that distance between detectors should be approximately 4 cm, has been done and a higher efficiency than putting the same number of detectors in each ring has been found. Results obtained are summarised in table 1. A configuration with 24/29/34/38 detectors respectively distributed in 4 rings was the one presenting both the best total neutron detection efficiency and the highest efficiency by detector.

Table 1 – Different number of detectors.

Configuration	Total number of detectors	Efficiency	Efficiency/Detectors
32 detectors / ring (4 rings)	128	1	0.78%
23 / 29 / 36 / 42 detectors (4 rings)	130	1.032	0.79%
26 / 30 / 34 / 38 detectors (4 rings)	128	1.050	0.82%
24 / 29 / 34 / 38 detectors (4 rings)	**125**	**1.052**	**0.84%**

If only polyethylene is used as moderator inside the whole structure of the counter, a Monte Carlo simulation shows that the average mfp in the portion of polyethylene surrounding the two first rings of detectors is approximately 1cm while it becomes 3.5cm if considering the slab containing rings number three and four. So, one important conclusion can be extracted from this results: since the mfp in the outer slab of polyethylene is approximately equal to the absorption mfp, it should be interesting to find another type of moderator to replace polyethylene surrounding outer rings (number three and four) in order to avoid a huge neutron absorption factor.

The final efficiency of the system and their distribution among the four rings of detectors are very sensible to detectors position (see table 2). It is better to have the maximum of the efficiency located at the second ring of detectors because it helps to guarantee a flat energy response of the detector.

Table 2 – How rings position alters total neutron detection efficiency.

Rings Position (cm)	Efficiency / Ring (%)	Efficiency / Detector (%)	Relative Efficiency
15.00	17.85 ± 0.09	0.744	1.0000 (0.26%)
19.00	19.88 ±0.09	0.686	
23.50	13.27 ± 0.07	0.390	
28.00	10.11 ± 0.06	0.266	
14.75	16.52 ± 0.09	0.688	**1.0064 (0.26%)**
18.75	20.78 ± 0.10	0.716	
23.25	13.73 ± 0.08	0.403	
27.75	10.47 ± 0.07	0.275	

3. Proportional counter gas composition.

It is well known that going to higher ^{3}He pressure a double effect is obtained: an increase in the efficiency of the system as well as the decrease of the die-away of the counter. The reason is that increasing ^{3}He pressures the detector becomes more and more sensitive to epi-thermal neutrons (see figure 3).

One of the initial requirements of this counter was that it should respect the theory developed by Hage [1], which is based on thermal neutrons dynamics, in order to have the guaranty that the analysis of the results could be done following this mathematical theory. When epi-thermal neutrons are absorbed, the dynamics of the system changes (it becomes faster) and a second exponential in addition to that predicted by the theoretical model is needed to fit the die-away of the counter. The best agreement between the MCNP calculated time function and the single exponential fit is observed at a ^{3}He pressure of 4 atm. Figure 4 shows the MCNP simulation results. The fit result for the calculated die-away time is 40.4 μs.

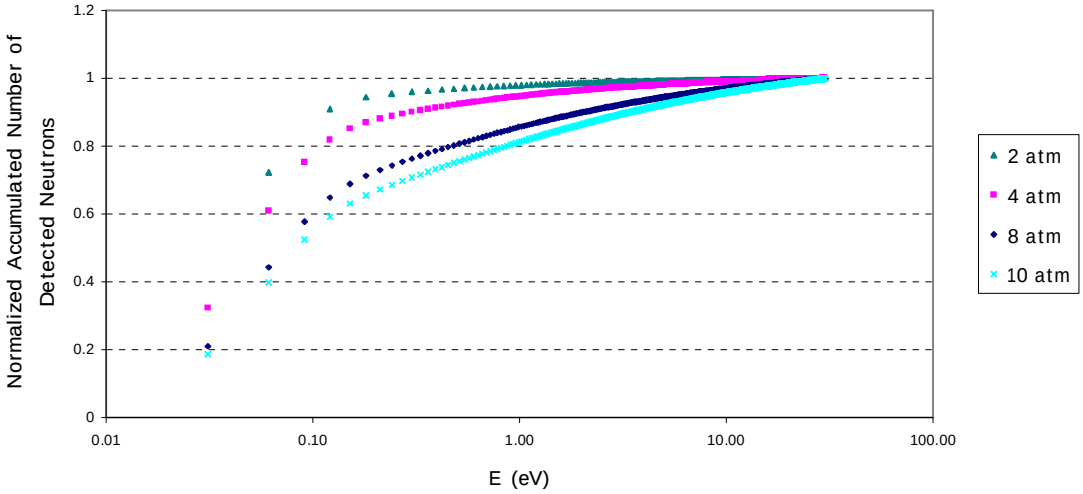

Figure 3 - Probability of detecting non-thermal neutrons as a function of the pressure.

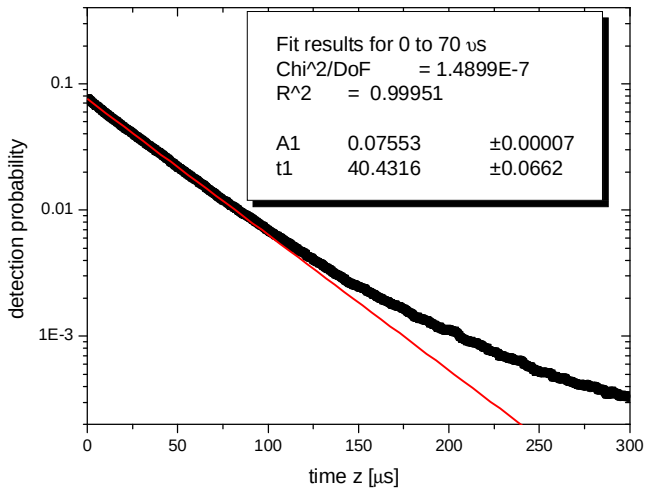

Figure 4 – Die-away calculation with 70μs of time interval acquisition.

However, the behaviour of proportional counters is not only depending on the ^{3}He pressure. The additional compounds fed into them to improve some of their qualities must be studied. A set of twenty tubes has been tested to decide which gas composition is the most adequate to the SNMC application. Two different pressures of ^{3}He have been taken into consideration: 4 and 8 atmospheres [3]. Three manufacturers have been contacted to have a general overview not only about the chemical composition, but also concerning the mechanical construction of the tubes that is essential to assure a good operation during manipulation and lifetime of the counter.

The proportional counter time response has been measured using a high resolution (20 ns bin width) Pulse Interval Analyzer (PIA) card in order to determine the dead time of the detectors [4]. Optimal HV working settings have been respected for each one of the ten different kinds of detectors under study. Dead time has been measured in very short shaping time conditions (250 ns). The double peak effect due to the discrimination of the protons and tritons tracks is only observed when there is no stopping gas inside the detector.

Figure 5 shows the differences in time response existing between those ten sorts of mixtures. It is possible to see that the ^{3}He pressure is not influencing that parameter, as the quantity and quality of stopping gas does it. The dead time decreases when Ar pressure increases. Xe is not so efficient as Ar in reducing the dead time of the counters.
The exact composition of detectors named X1 and X2 is unknown. The only explanation received was that one of them had been produced to work in high-count rates conditions and the other one was thought to have a long life. Analysing the results presented in figure 6 it results easy to see that X1 is adapted to be working with high-count rates and X2 will be the one having a long life.

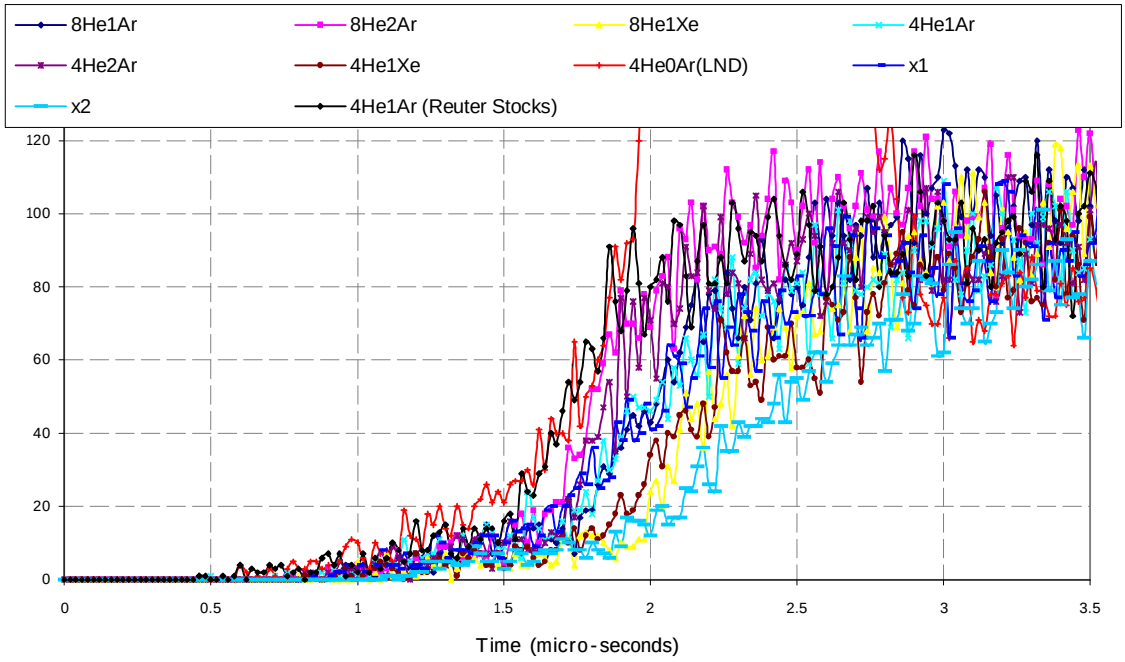

Figure 5 – Proportional counters time response obtained with a high resolution (20 ns bin width) Pulse Interval Analyzer (PIA).

As the main goal of this counter is to detect the maximum number of neutrons coming from the spontaneous fission in order to have a good statistic for triplets, a fast response from the proportional counters is required.

Figure 6 illustrates how the presence of stopping gas changes the operational HV. Obviously the final choice was to work with detectors containing 4 atmospheres of ^{3}He plus 2 atmospheres of Ar because of both their fast response.

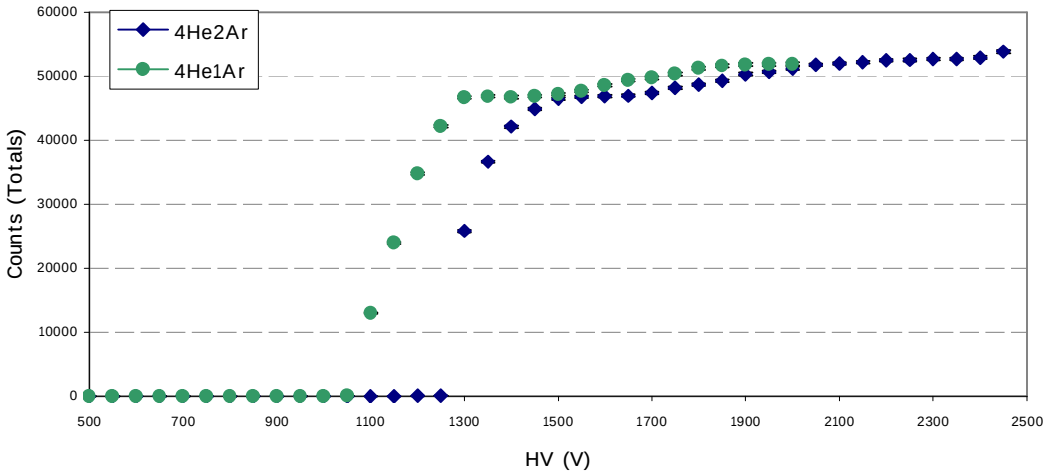

Figure 6 – Working plateau for proportional counters filled in with 4 atm of ^{3}He plus 1 or 2 atm of Ar.

4. Energy response.

A counter whose counting efficiency does not depend on the neutron energy can be a very useful device in many areas of neutron physics. For an ideal counter of this type, a graph of the detection efficiency versus neutron energy is a horizontal line, which has led to the name flat response counters. To this purpose horizontal Cd-liners (1 mm thick) were added delimitating the active length of the detectors. Although no real counter exists with a perfectly flat response over the entire range of possible neutron energies, several designs have evolved that come close to this ideal.

A set of 54 MCNP calculations has been done with mono-energetic neutrons ranging from 0.4 eV to 5 MeV. Results obtained are showed in figure 7 where are compared with other two counters developed at Los Alamos National Laboratory with similar characteristics. As figure below illustrates, the behaviour of the SNMC is very similar to those corresponding to the Plutonium Scrap or the Pyrochemical Counters, however the new SNMC presents the highest efficiency. It is also possible to see the efficiency distribution around the four rings of detectors forming the SNMC.

Multiplicity counters achieve their flat energy response largely through the use of multiple rings of Helium-3 placed at different depths in the moderator material. Figure 7 plots the relative count rate responses for the four tube rings in the SNMC as a function of neutrons energy. Each ring responds differently, but the sum of all four, as plotted in figures above is nearly constant.

In figure 9 two absorption peaks are observed. The first one, situated around 100eV, corresponds to the absorption in Cd. The second one, situated at around 3 MeV, comes from the absorption in Graphite.

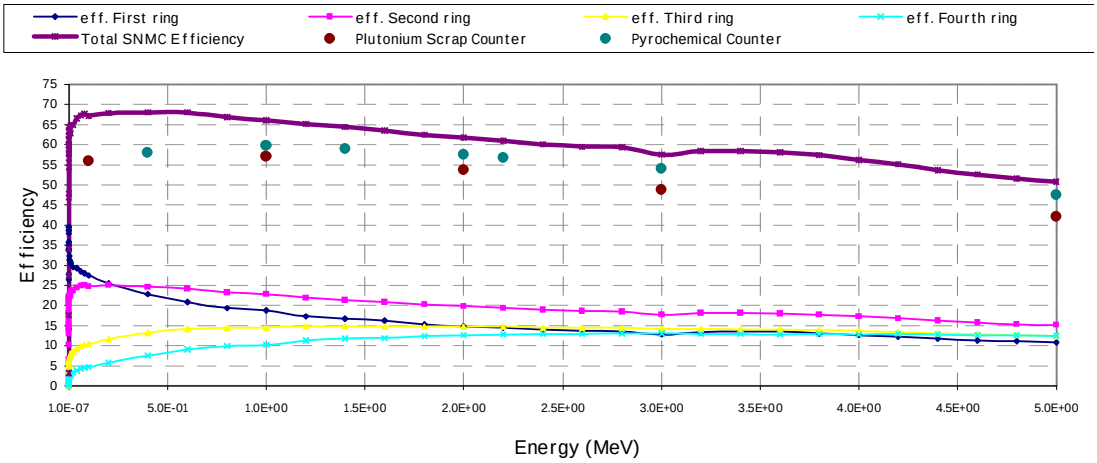

Figure 7 – Energy Response of the SNMC compared with The Plutonium Scrap and the Pyrochemical counters.

5. Vertical profile.

The final dimension of the assay chamber (80cm length) allows allocating without problems two or more samples reducing in this way the acquisition time in order to make agile the process of inspection of the nuclear sites. The axial efficiency profile has been simulated by MCNP calculations using a punctual ^{252}Cf point source situated along the vertical axis of the cavity. Results obtained are summarised in figure 8.

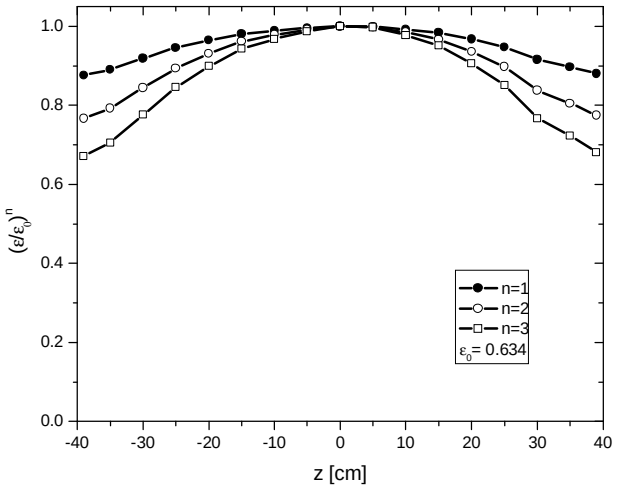

Figure 8 – Normalised powers of the detection efficiency calculated with MCNP at several position along the z-axis of the sample cavity of the SNMC.

The axial efficiency profile varies less than ±2% over a total height of the cavity of 50 cm. The total variation over the complete height of the sample cavity is less than ±5%. Of course, the integral response for a can of Pu has less variation than the measured ^{252}Cf point source.

6. Horizontal profile.

The same ^{252}Cf source was used to measure the radial response variation at the midplane of the sample cavity. A total of 375 points have been simulated along the cavity and results obtained are

represented in figure 9. The mean neutron detection efficiency and its associated error, calculated as the standard deviation of the 375 points studied, are (63.7±0.3)%. In general, the sample should be centred in the cavity and the integral radial variation will be minor than 0.3%.

7. Structure of the Junction Box

The signals from the 125 ^{3}He detectors are transferred to a total of 44 amplifiers. The amplifier output signals are fed into 6 Multi Input- Pulse Train Analyser (MI-PTA) MIXERs (each with 8 inputs). In the case of the SNMC there are 5 SLAVE MIXERs each one connected to 8 amplifiers and a MASTER MIXER who is connected to the 4 residual amplifiers. The MASTER MIXER controls the SLAVE MIXERs, it collects the counts from the SLAVE MIXERs, and communicates with a PC via a USB 2.0 port. It also controls the 2 High Voltage supplies, and monitors the voltages and the system temperature. The complete neutron multiplicity analysis is performed real-time on a PC.

The cylindrical junction box has been split in two horizontal layers (see figure 10). The bottom level contains a double layer printed circuit (PCB) that establishes the connection between the different groups of tubes, distributes the HV around the different rings of detectors, and guarantees the ground connection to each ^{3}He detector (see figure 11). The output signal collected by each group of detectors is transferred through a Lemo connector to the amplifiers, which are situated in the upper level. These connectors have been fixed at the PCB. The MIXERs are also placed on the second level, each in a Faraday box to reduce electromagnetic emissions, which could disturb the amplifiers. Two High Voltage modules are situated always in the second level of the junction box.

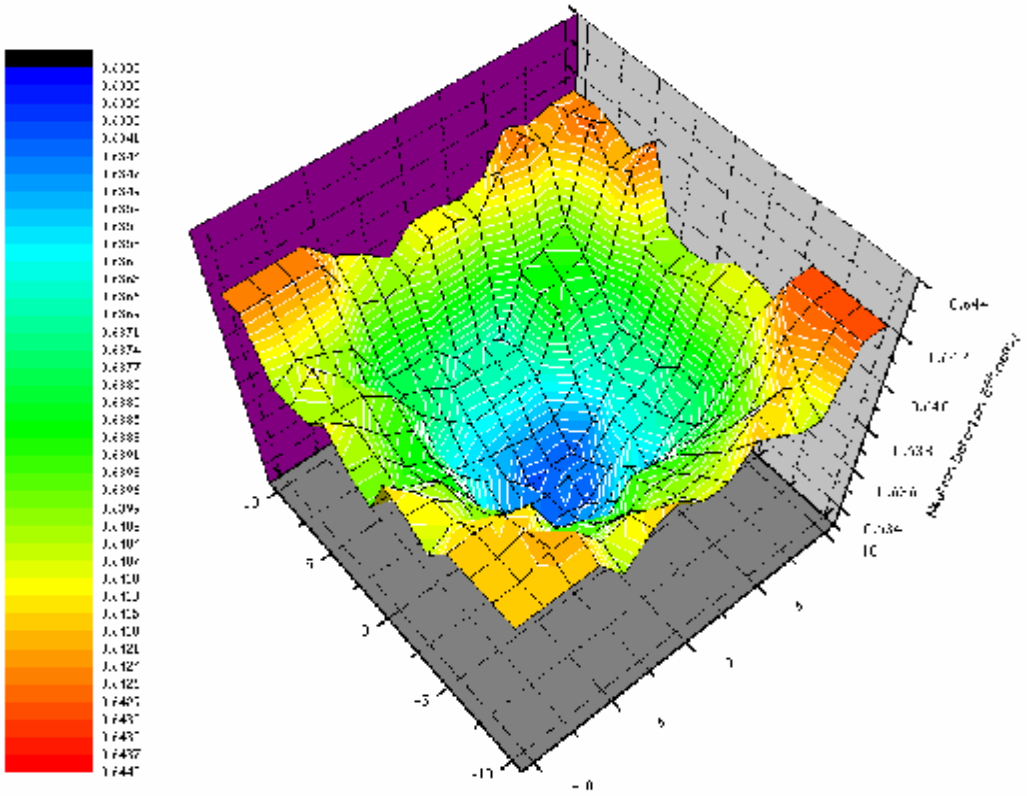

Figure 9 – Radial response variation at a mid-plane of the sample cavity.

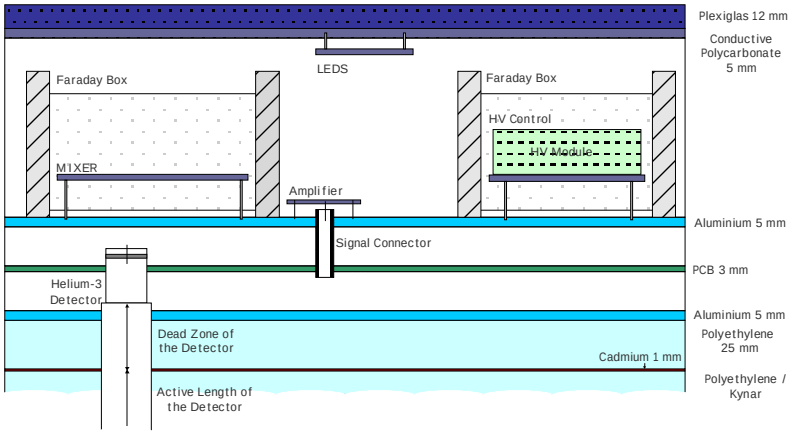

Figure 10 – Vertical section of the junction box.

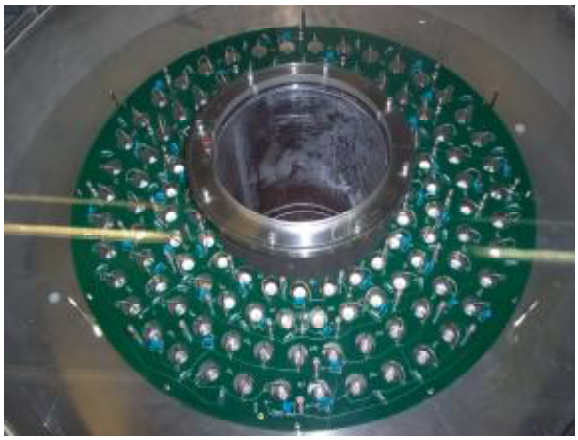

Figure 11 – Printed circuit (PCB) with RC filters to each group, HV decoupling capacitors, ground connections and the signal connectors (I).

8. References

[1] Cifarelli D.M. and Hage W.; *Models for a Three Parameter Analysis of Neutron Signal Correlation Measurements for Fissile Material Assay; Nucl. Instrum. Methods*; 1986; A251; p 550.

[2] Dabrowska J., Drozdowicz K., and Krynicka E.; *Decay of the pulsed thermal neutron flux in polyethylene samples. Experimental and MCNP simulation; Report No. 1861/PN*; The work has been performed in the frame of the Project No. 9 T12B 027 16 of the State Committee for Scientific Research; INSTITUTE OF NUCLEAR PHYSICS Ul. Radzikowskiego 152, 31-342 Kraków, Poland.

[3] Peerani P., Marín Ferrer M., Ravazzani A.; *Study on the Influence of ^{3}He Pressure on the Efficiency of Neutron Collars; European Commission - Joint Research Centre, Institute for the Protection and the Security of the Citizen. Non-Proliferation and Nuclear Safeguards Unit. Technical Note No. I.02.66.*; July 2002.

[4] Looman M., Marín Ferrer M., Pedersen B., Peerani P.; *Novel Components for Neutron Counters; 25th Symposium on safeguards and nuclear materials management (ESARDA)*; 2003,Stockholm.

[5] Evans R.D.; *The atomic nucleus; McGraw-Hill Book Co.;* 1955; New York.

[6] Looman M.R., Peerani P., and Schillebeeckx P.; *An Overview of NDA Instruments Modeled with the MCNP-PTA Code at the JRC Ispra; 23rd Annual Meeting: Symposium on Safeguards and Nuclear Material Management*; 2001; Report EUR 19944 EN.

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Monte Carlo Modeling of a Fork Detector System

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Monte Carlo Modeling of a Fork Detector System

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Abstract:

Monte Carlo simulations are described for the response of a Fork detector system used to estimate neutron and gamma count rates from a spent-fuel assembly. Using spent-fuel source-strength characterizations from the Oak Ridge National Laboratory (ORNL) Standardized Computer Analyses for Licensing Evaluation (SCALE) system and the Monte Carlo Neutron-Particle (MCNP) code, Fork detector responses were estimated by starting with the fuel characterization and burnup history of an assembly. Detector response to neutrons and photons is described for pressurized-water-reactor (PWR) and boiling-water-reactor (BWR) systems. Particle fluxes are calculated, and detector signals are then estimated. Some preliminary benchmarking is discussed using Fork measurement data from a small number of spent-fuel assemblies.

Keywords: Monte Carlo; Fork detectors; spent fuel

1. Introduction

Fork radiation detectors[1] are commonly used to monitor spent-nuclear-fuel assemblies. They provide a convenient nondestructive means for characterizing fuel assemblies following discharge. These Fork detectors are physically located outside the assembly and provide information on the neutron and gamma radiation fields emanating from the spent fuel. In safeguards applications, it is of interest to determine if such a system can detect anomalies in spent-fuel assemblies (e.g., missing fuel rods); another application of the Fork detector would be the verification of assembly burnup. The response of a Fork detector is complicated, and the interpretation of detector response for a particular assembly configuration requires careful analysis.

ORNL maintains and develops the ORNL SCALE[2] system with many applications to reactor analysis and related areas. One prominent application of SCALE is for the analysis of fuel-assembly burnup and the determination of the characteristics of spent fuel. In an effort to proceed from the characterization of spent fuel to the determination of Fork detector response, we developed a Monte Carlo transport model to simulate detector response by starting with the spent-fuel-assembly source composition. This Monte Carlo approach is based on the MCNP code[3]. In this paper we describe the MCNP model leading to the determination of Fork detector response. The ORNL SCALE system is used to determine the MCNP source strengths that drive the transport calculations. We also discuss preliminary benchmarking studies that compare calculated results against neutron and gamma measurements performed by the European Atomic Energy Community (EURATOM).

2. MCNP models of detectors and assemblies

The MCNP model discussed here consists of a Fork detector plus a spent-fuel assembly. Currently, the model exists in two versions, which treat BWR and PWR fuel assemblies. The MCNP model includes the assembly (BWR or PWR with fuel rods and water rods, as the case may be) and two Fork

[a] Managed by UT-Battelle, LLC, for the U.S. Department of Energy under contract No. DE-AC05-00OR22725

detector arms mounted such that one is on each side of the assembly. Generally, because measurements are performed in the cooling pool, this entire system is immersed in water. The cooling-pool water may also contain boron. For the PWR cases studied, the model has 2200 ppm of boron in the water. In the BWR cases, the water was boron free. These particular boron concentration values were chosen because they apply to the cases used for benchmarking. The data on Fork detector design and performance that are used here, as well as the data used for benchmarking, were obtained from EURATOM[4].

The example used for the PWR case involves a Westinghouse 17 × 17 PWR fuel assembly containing 264 fuel rods. The BWR case contains an 8 × 8 assembly. The 8 × 8 assembly has 1 water rod and 63 fuel rods. The material composition of the fuel rods was that of fuel irradiated to 40 GWd/t. However, it contained just the actinide component; fission products were not included.

Each arm of the Fork detector contains two fission chambers for neutron detection and an ionization chamber for gamma detection. One of the fission chambers is surrounded by high-density polyethylene (HDPE), which is covered with a 0.5 -mm cadmium shield. The other fission chamber is uncovered, although it is inside a 4-mm-thick cylindrical shell of HDPE that surrounds all of the detectors. The cadmium reduces the thermal neutron flux from the pool, and the HDPE then thermalizes the faster neutrons that penetrate the cadmium. In the descriptions that follow, the chamber to the inside of the cadmium shield will be referred to as the shielded chamber and the other chamber will be referred to as the unshielded chamber. The ionization chamber is encased in a cylindrical brass tube with a 2-mm wall thickness and is surrounded by the HDPE and cadmium layers.

The Fork detector arms are placed on each side of the assembly with the assembly located midway between them. Because of the different widths of BWR and PWR assemblies, the separation distance between the arms is different in the two cases. The inside distances between the arms are 16.8 and 24.3 cm in the BWR and PWR cases, respectively. Figure 1 is a vertical cross section showing the two detector arms with an 8 × 8 assembly located between them. Two fission chambers and one ionization chamber (surrounded by a brass sleeve) can be seen in each arm. The detector arms are the same but are rotated by 180 degrees relative to one another. Fork detector arms have a diameter of 9.2 cm.

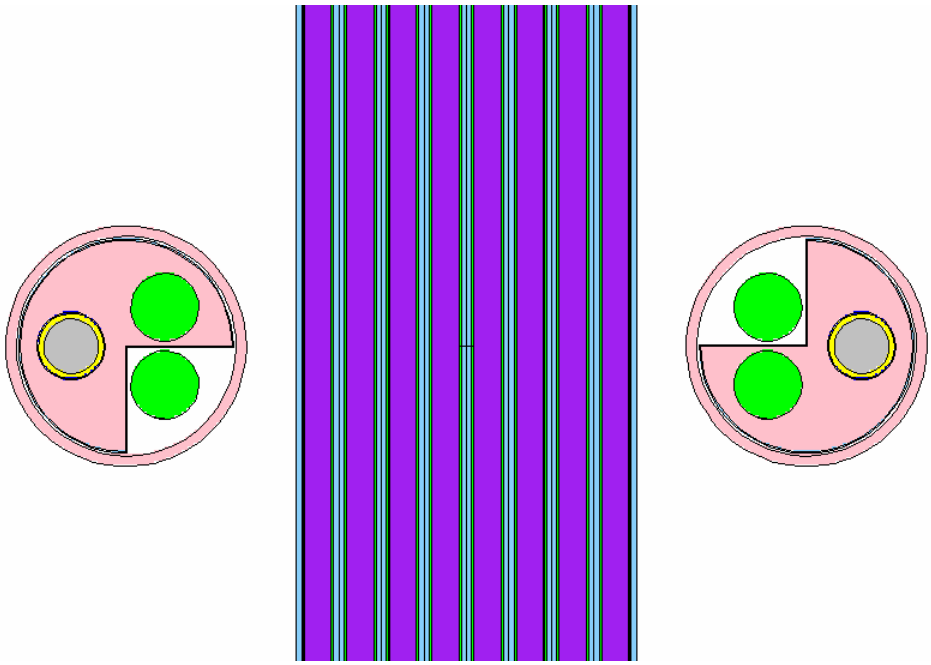

Figure 1. Vertical cross section through the Fork detector arms with an 8 × 8 BWR assembly. For the arm on the right of the figure, the two fission chambers are to the left, with the bottom one shielded and the top one unshielded. The ionization chamber is to the right. Both arms are the same but are rotated by 180 degrees around the horizontal axis relative to one another.

3. Detector response estimation

MCNP calculations result in particle flux estimates. The fluxes are calculated as a function of energy. The gamma detector (an ionization chamber) has a response that is a function of exposure. If one determines the photon flux as a function of energy, one can then determine exposure via a conversion factor that is a function of energy. The fission chambers register counts that correspond to fission events occurring in the wall coating which is enriched in ^{235}U. Fission chambers are mostly sensitive to thermal neutrons; however, for accurate calculation of the total response, one needs to determine neutron flux as a function of energy and then use the energy-dependent fission cross section to determine fission rates in the chamber.

The radiation sources are modeled within each spent-fuel rod, with the total source strength determined using the concentration of the various nuclide species in the fuel at the measurement time of interest (e.g., ^{137}Cs, an important gamma emitter, and ^{244}Cm, an important source of neutrons). Source strength, in general, may vary as a function of position within the assembly. MCNP is capable of defining complicated source configurations; axial and horizontal variations in source strength can easily be accommodated. MCNP also allows flexibility in the specification of the source-energy spectrum. Source-energy spectra are output from SCALE codes that simulate assembly burnup and cooling. Energy-bin boundaries can be determined by the user, and MCNP can accept any arbitrary set of bin boundaries.

4. Preliminary efficiency and normalization studies

In the initial simulation runs, overall efficiencies of the 17×17 and 8×8 models were assessed. As might be expected, for a given number of histories, the 8×8 case executes more quickly. (There are fewer cells and the geometry is simpler, giving rise to less complicated trajectories.) Neutron-tracking cases take considerably longer than photon-tracking cases. From early test results, it is also apparent (especially for gamma sources) that the internal fuel rods make negligible contributions to the detector counts. So, for purposes of efficiency one may apply biasing techniques to lower the importance of the internal rods.

There is one gamma-counting device (an ionization chamber) in each arm of the Fork detector and, for neutron detection, there is one shielded and one unshielded fission chamber in each arm. The ionization chamber produces a signal that is proportional to dose rate. The Fork detector ionization chambers are quoted by the manufacturer as having a response of 3.4×10^{-10} A/R/h. Because the transport calculations result in estimates of photon flux, it is necessary to convert calculated photon fluxes to dose rates using flux-to-dose-rate conversion factors. Then, via the response function of the ionization chamber, the magnitude of the ionization chamber gamma signal can be determined. The flux-to-dose-rate conversion factors are functions of energy, and the photon flux tallies from MCNP are scored in energy bins.

The fission chambers in the Fork detector have a thin layer of UO_2 on the inside cylindrical surface. This layer is enriched to 93% in ^{235}U so that an effective surface density of 850 μg/cm^2 exists for ^{235}U. The quoted detector response is 0.14 cps/nv, which is to be interpreted as counts per second per unit of flux. (We assume this value applies to thermal neutrons.) Figure 2 shows the results of an MCNP simulation for a layer of UO_2 with the design concentration of ^{235}U on the inside of one of the fission chambers. The fission rate can be seen to reproduce the quoted chamber response value for thermal neutrons. (Presumably, the chamber response does not account for self-absorption of fission fragments.) In determining the response of a neutron detector, the energy-dependent neutron flux, together with the energy-dependent fission cross section, is used to estimate the fission rate in the UO_2 on the inside of the chamber wall.

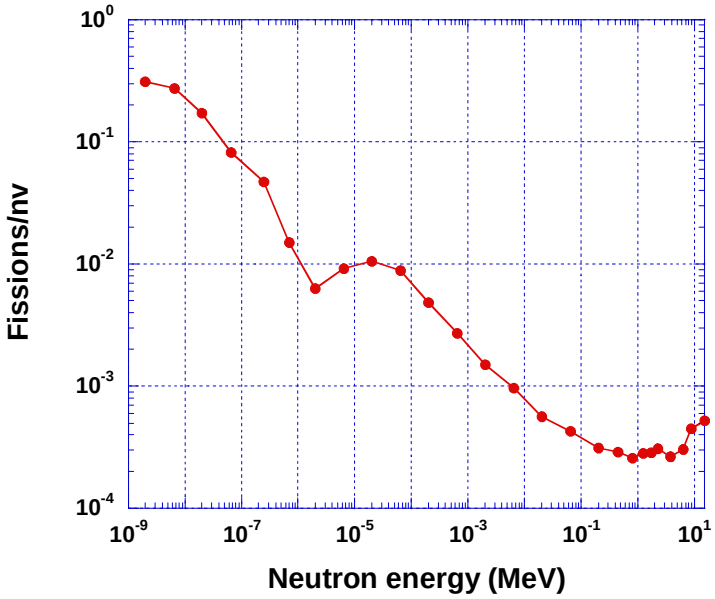

Figure 2. Fission rate versus neutron energy for the fission chamber. This has the form of the ^{235}U fission cross section and reproduces the quoted response of 0.14 cps/nv at 0.025 eV.

5. Neutron and gamma response functions

Fork detector responses to neutrons and photons have been estimated as a function of the originating neutron or photon energy. This was done for both a 17×17 PWR and an 8×8 BWR. We will later refer to an 18×18 PWR case and a 9×9 BWR case when discussing benchmarking studies. The response in the 9×9 case is similar to that of the 8×8, and the 17×17 can be considered as representative of an 18×18. The various calculated results will be presented versus the energy of the originating neutron or photon. These results are per neutron or photon for the average neutron or photon originating in the assembly. The calculations assume that the radiation source within the assembly is distributed among the fuel rods in a horizontally uniform manner. However, the source strength has an axial profile, which is a typical assembly axial profile. All measurements and simulations will be for the assembly centerline.

Spent-fuel composition and the source strength are independently specified in the MCNP model. The spent-fuel composition is intended to be representative of a discharged assembly. Because all significant actinides are included in the composition, the Monte Carlo calculations account for secondary fission neutrons. However, the fuel assembly model did not contain neutron poisons.

6. Neutron response results

Neutron results are presented both as neutron flux at the location of the two fission chambers (the shielded and unshielded chambers) and as count rates in these detectors. Because of the symmetrical arrangement of the Fork detector arms, the response in a particular type of detector will be the same for both arms of the detector (within the statistical limits of the stochastic Monte Carlo process). Most results reported here were obtained by calculating an average for the two detector arms; but all quoted values are per detector. The flux values (neutrons/cm^2/s) are averages for the region occupied by the active part of the fission chamber. The count rates are determined from the

flux, the ^{235}U fission cross section, the ^{235}U density, and the volume of the UO_2 layer that coats the inside surface of the fission-chamber wall. The reaction rate, *R*, is calculated using the thin-target approximation, as in

$$R = Nn\sigma d$$

where *N* is the number of incident particles/s, *n* is the density of ^{235}U nuclei, and *d* is the thickness of the UO_2 layer. Note that the fission rates are calculated from the average flux in the region occupied by the UO_2 layer, whereas the flux values that are quoted are averages over the full active volume of the fission chamber. However, the flux in both regions is essentially the same.

In the figures that follow, neutron fluxes are presented both as total flux and as flux below 0.4 eV. The flux below 0.4 eV will be referred to as the thermal flux (the flux below the cadmium cutoff). The flux values (total and thermal) and the count rates are presented as functions of the energy of the neutron originating in the assembly. Figures 3–6 show various neutron simulation results. Total flux per originating neutron in the 17×17 assembly is shown in Figure 3. Results are shown for both the shielded and unshielded fission chambers. Figure 4 shows total flux from the 8×8 assembly. The 8×8 values are higher, probably reflecting the fact that the average neutron originating in that assembly is nearer to the detector than is the case for the 17×17 assembly. (Note also the different relative spacing of the shielded and unshielded values in the PWR and BWR cases; see below for more discussion of this.) These neutron fluxes produce counts in the fission chambers. Figure 5 shows the resulting count rates in both the shielded and unshielded fission chambers for the 8×8 BWR case (per originating neutron). The fission chamber is mostly responsive to thermal neutrons, as seen in the thermal neutron flux (flux below 0.4 eV) for the 8×8 case shown in Figure 6. By comparison of Figures 5 and 6, one can see that the fission-chamber signal is driven by the thermal flux. For the chamber count-rate curves, we show least-squares fits, whereas for the flux plots, the calculated points are simply connected with straight lines. In general, the scatter seen in the calculated points results from the stochastic nature of the Monte Carlo calculations. It should, however, be noted that a real depression in flux, caused by a resonance in ^{16}O, is seen around 5 MeV.

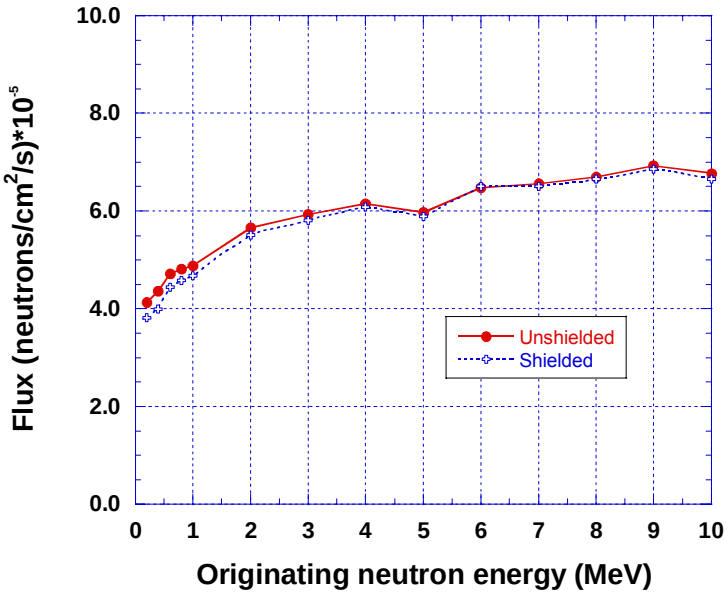

Figure 3. Total neutron flux (i.e., per originating assembly neutron per second) at the fission chambers as a function of originating neutron energy for a 17 × 17 PWR assembly. The calculated values have been joined by straight lines. Although there is some statistical scatter in the results, the depression around 5 MeV is real and is caused by a resonance in ^{16}O.

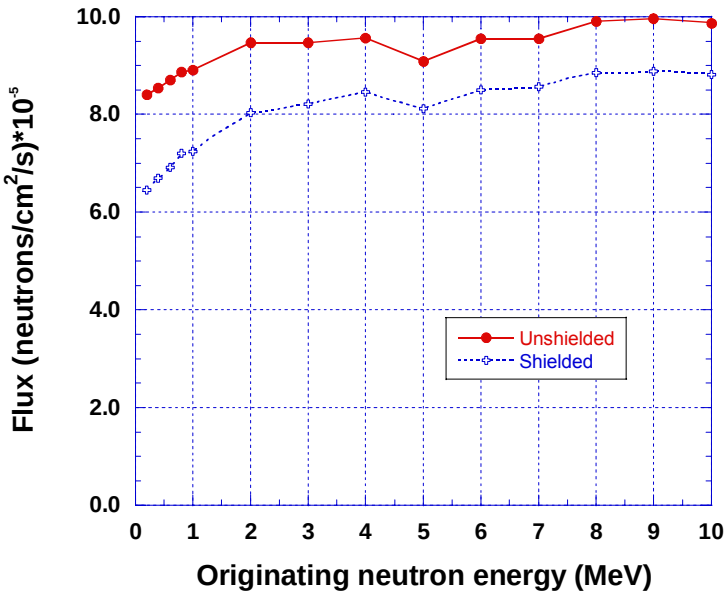

Figure 4. Total neutron flux (i.e., per originating assembly neutron per second) at the fission chamber versus neutron energy for the 8 × 8 BWR assembly. Again, note the flux depression around 5 MeV.

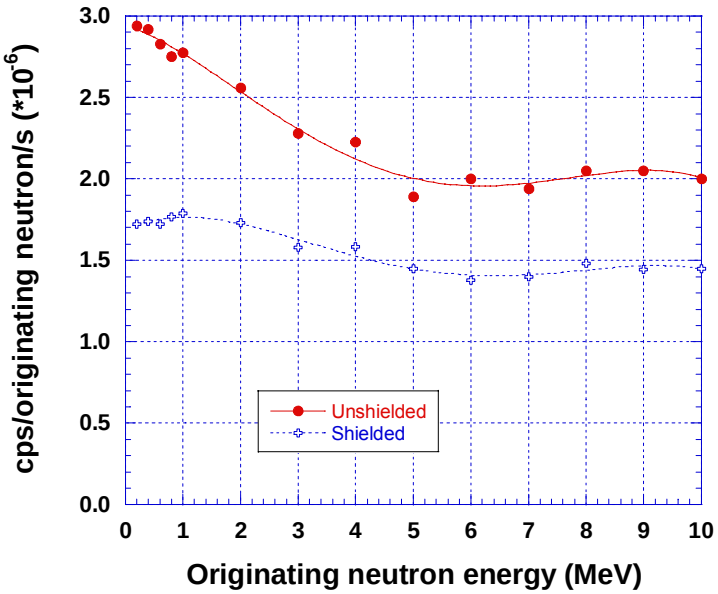

Figure 5. Fission-chamber count-rate response for the 8 × 8 BWR case. The points are the calculated values. They include statistical scatter and have been fitted with smooth curves.

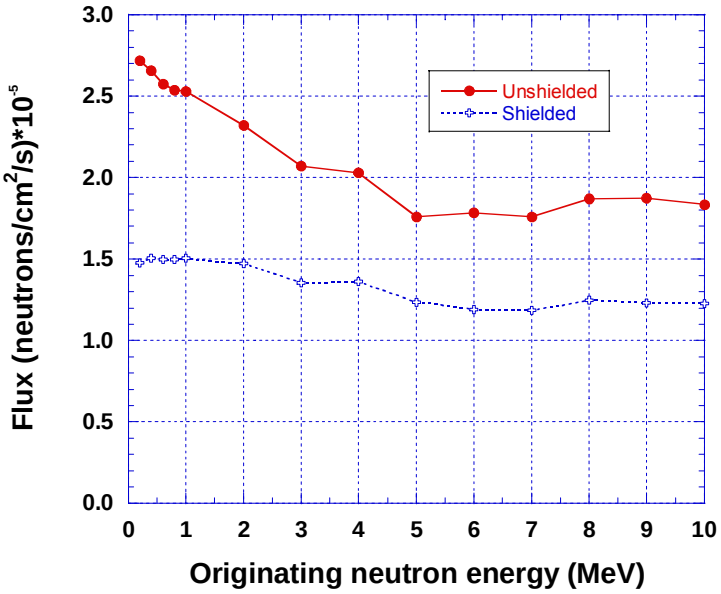

Figure 6. The thermal (< 0.4 eV) neutron flux at the fission chambers for the 8 × 8 BWR case.

Figure 7 shows the fission-chamber count-rate response (shielded and unshielded) in the case of the 17 × 17 Fork configuration. This figure should be compared with Figure 5. As was noted, it is mostly thermal neutrons that contribute to the fission-chamber response. Therefore, keeping in mind that the PWR pool contains boron and the BWR pool does not, fewer thermal neutrons will reach the vicinity of the detectors in the PWR case. Consequently, the unshielded detector will see less thermal neutrons from the PWR, thus leading to a lower signal, as is evident from the figures. (This is also reflected in the unshielded flux being relatively lower in Figure 3 as compared with than in Figure 4.)

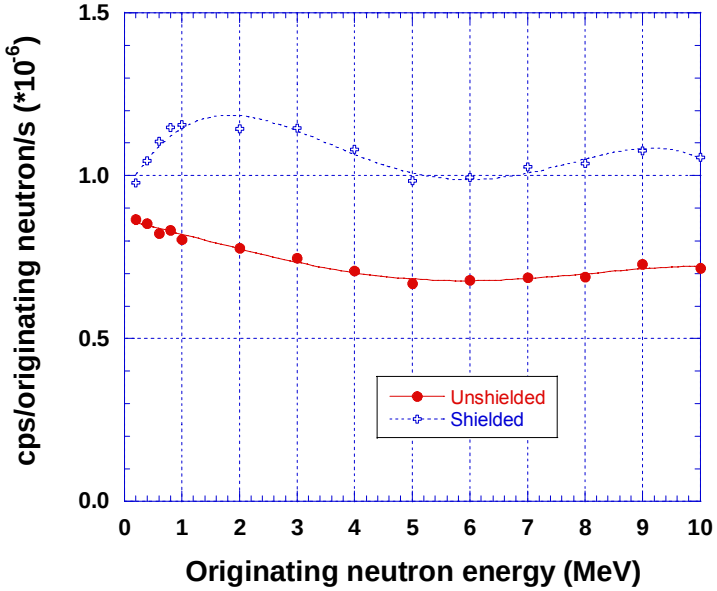

Figure 7. The fission-chamber count-rate response for the 17 × 17 PWR case. The calculated points have been fitted with smooth curves.

The shielded detectors are expected to detect fast neutrons that pass through the cadmium and are then thermalized in the HDPE. This neutron population should not be greatly influenced by the boron and therefore should be similar for the BWR and PWR cases. However, note that the average neutron originating in the PWR assembly is farther from the detectors than in the BWR case (and is subject to more shielding). So, although we expect rough agreement between shielded responses (per originating neutron) for the BWR and PWR, the PWR response should be somewhat lower. Again, this is evident from the figures.

The count-rate response curves show a fairly flat energy response, and, furthermore, the shape of the neutron energy spectrum does not vary appreciably from case to case. Therefore, it was not surprising to find that a standard form of neutron energy spectrum seemed to suffice for all calculations. Thus, we have chosen to use the ^{244}Cm neutron decay spectrum for all calculations.

Table 1 summarizes MCNP-predicted fission rates at the chamber locations when the neutron spectrum is that of a ^{244}Cm source. Specifically, the quantities in Table 1 are fissions per cubic centimeter, in the UO_2 layer, per originating neutron.

	BWR	PWR
Unshielded	2.36×10^{-4}	7.45×10^{-5}
Shielded	1.59×10^{-4}	1.032×10^{-4}

Table 1. Summary of MCNP fission rates from a ^{244}Cm source (fissions per cubic centimeter).

7. Gamma response results

The gamma results are presented in two ways. We show both the photon flux and the (hypothetical) gamma dose rate over the active part of the ionization chamber. Again, the results reported are per originating photon and are shown as a function of the originating photon energy. The primary quantity that is calculated is the photon flux versus energy at the chamber location. The ionization properties of the radiation field are probably best described in terms of the exposure. However, flux-to-dose-rate conversion factors are readily available and these were employed to convert the gamma flux to the dose rate. The chamber sensitivity is quoted in terms of dose, and the resulting signal can then be calculated.

The photon flux for the 8×8 case is illustrated in Figure 8. This is the total flux at the ionization chamber per originating photon per second. The shape of the curve shows that low-energy photons are less likely to escape from the assembly, because of photoelectric capture in the high-Z fuel material. The resulting photon dose is shown in Figure 9. Results from two separate dose-rate-response functions are shown; one set of results was obtained using American National Standards Institute/American Nuclear Society (ANSI/ANS) flux-to-dose-rate conversion factors[5], and the other set was obtained using International Commission on Radiological Protection (ICRP) conversion factors[6]. The results for the 17×17 PWR case are quite similar.

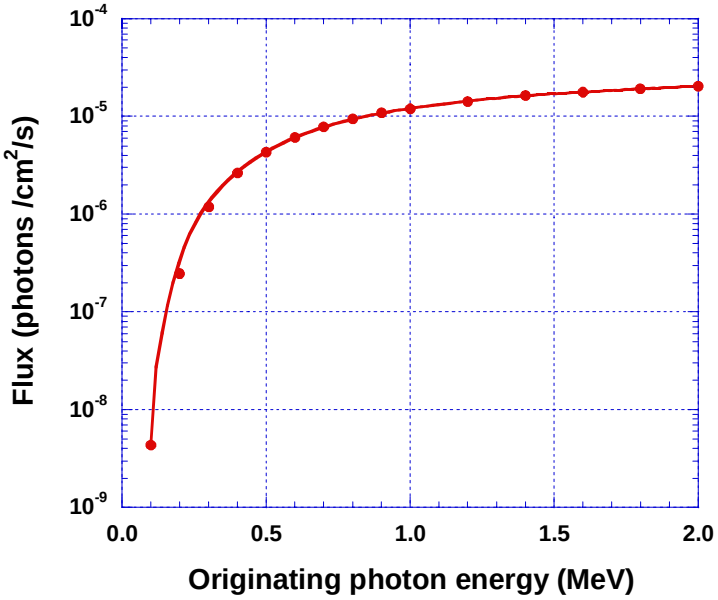

Figure 8. Photon flux (i.e., per originating photon per second) at the ionization chamber for the 8 × 8 BWR case. This is shown as a function of originating photon energy.

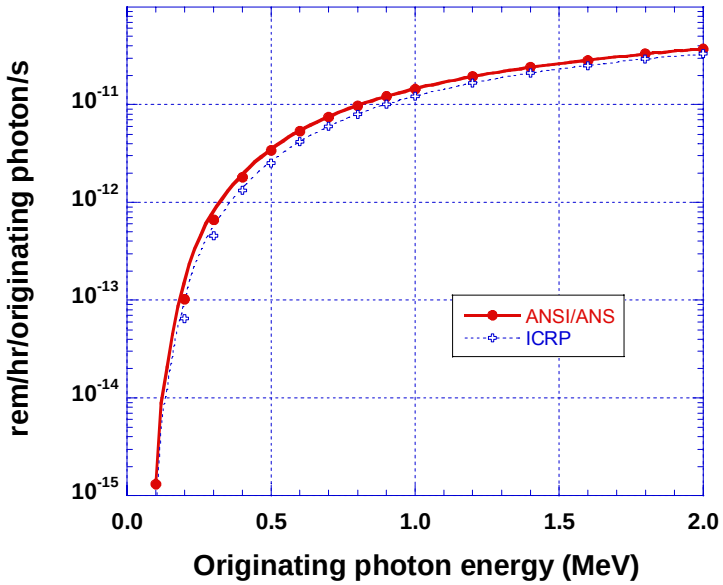

Figure 9. Photon dose results for the 8 × 8 BWR case as a function of originating photon energy. These values are dose rate (per hour) per originating photon rate (per second). Values calculated with both sets of flux-to-dose-rate conversion factors are shown.

8. Benchmarking of Fork simulations

Measurement data obtained by EURATOM were used to conduct benchmarking studies to see how well SCALE and MCNP reproduce the observed counts. Ideally, if trends versus burnup are to be readily detectable, it is desirable that all measurements be made on fuels that have the same initial enrichment and have experienced the same cooling time. In checking measurements reported for Fork detectors, one finds that most groups of measurements cover short burnup ranges (i.e., < 5,000 MWd/t). Because of scatter in the measurements, longer burnup ranges (e.g., > 10,000 MWd/t) may be needed in order to estimate trends.

Two groups of measurement data were examined. One group was for assemblies from a 9 × 9 BWR, and one was for assemblies from an 18 × 18 PWR. An 8 × 8 model and a 17 × 17 model were used to simulate the BWR and PWR assemblies, respectively, with neutron and gamma source strengths determined from the SAS2/ORIGEN-S sequences in the SCALE system. (The overall dimensions of the BWR and PWR modeled assemblies were essentially the same as the measured ones.)

9. Neutron results

Burnup for the 16 BWR assemblies studied ranged from about 29 to 32.4 GWd/t. All had an initial enrichment of approximately 2.6%. (For proprietary reasons, quoted numbers are rounded values, and the specific assemblies are not identified.) Measured data points are shown in Figure 10 for neutron counters A and B (shielded and unshielded, respectively). Calculated values are also shown for both the shielded and unshielded cases. The measured data points show scatter. Second-degree polynomial fits are shown for both sets. From Figure 10 we note that neutron A (shielded) counts are lower than the neutron B (unshielded) counts and that for the calculated values the shielded count rates are also lower than the unshielded ones. As explained, this should be the case for assemblies that do not have boron in the cooling-pool water. However, the trends as a function of burnup are not the same for measured and calculated count rates. (The measured values seem to increase more strongly with burnup.) With the amount of spread and the burnup range, one cannot be definitive about the trends in the measured data.

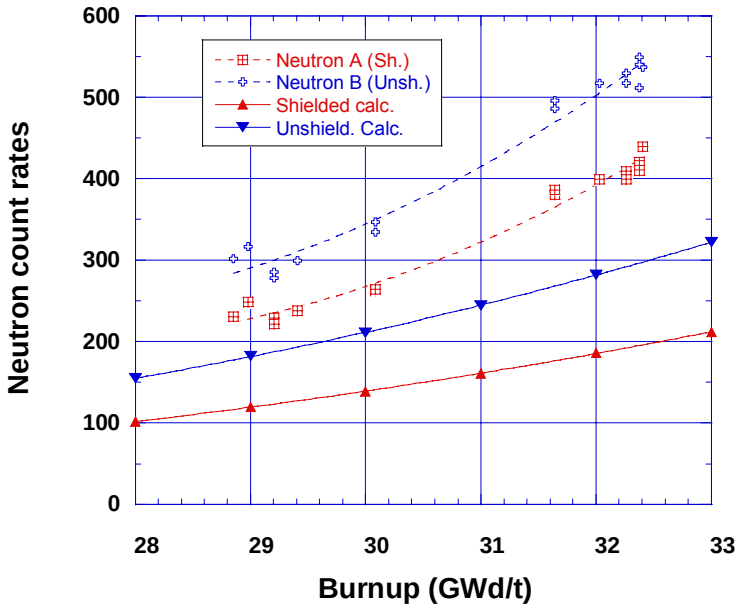

Figure 10. Neutron results for the BWR assemblies. The cooling pool does not contain boron, and unshielded rates are higher than shielded rates. Trends with burnup are different for the measured and the calculated cases. Second-degree polynomial fits are shown for the measured and calculated values.

The neutron results for the second set of assemblies (18 × 18 PWR) are shown in Figure 11. This set consists of five assemblies with initial enrichments of approximately 3.4% and with burnups ranging from about 34 to 41.6 GWd/t. Because of variations in enrichment, cooling time, and out-of-reactor time between irradiation cycles, we do not expect a smooth trend as a function of burnup. Again, estimated count rates are lower than measured ones. For these PWR assemblies there is boron in the pool water (2200 ppm) and, for both the measured and calculated points, the values for shielded count rates are higher. From Figure 11 one can see that the trend in the measured values is reproduced quite well by the simulations.

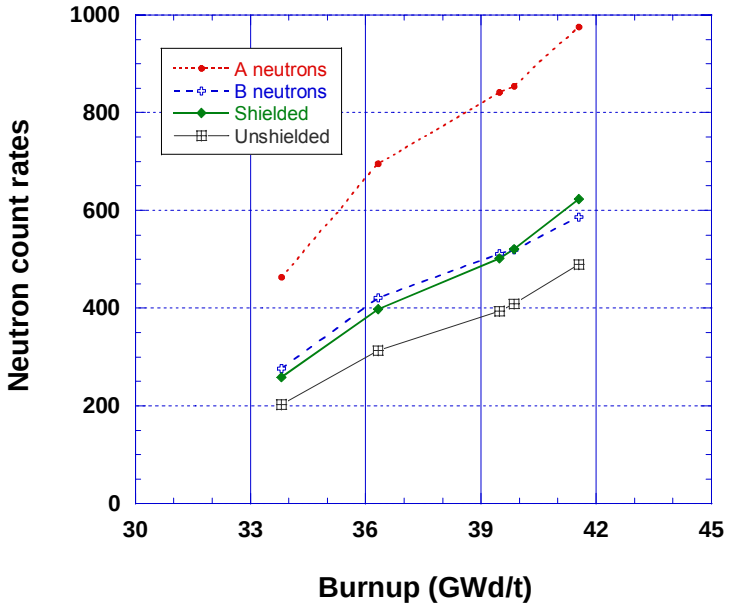

Figure 11. Neutron results for the PWR assemblies. The cooling pool contains 2200 ppm of boron, and shielded rates are higher than unshielded rates. Trends with burnup are similar for the measured and the calculated cases.

In summary, the neutron comparisons show two notable characteristics. (1) The observed count rates in the neutron detectors are higher than what are predicted. (2) Although measured and calculated trends as a function of burnup are similar in the PWR case, they are not so in the BWR case. A couple of possibilities come immediately to mind for the higher count rates: It may be that the active region of the fission chamber is larger than what was used in the simulations, or perhaps the fission chambers are affected by the gamma flux. (The gamma flux immediately outside a spent-fuel assembly can be as much as seven orders of magnitude higher than the neutron flux.) However, it is not surprising that count rates do not match exactly. We expect that in a practical application of this work, a normalization factor would be required.

One likely explanation for the lower calculated neutron count rates is that estimates of neutron source strength for the spent-fuel assemblies are low. The majority of the neutrons are from ^{244}Cm. The ^{244}Cm concentration is determined using reactor-burnup-simulation codes (from the SCALE system) and is subject to some degree of uncertainty. Work performed at ORNL[7] comparing simulations with destructive-analysis measurements indicates that ^{244}Cm concentrations seem to be underpredicted. Before addressing this possible underprediction, however, we first wish to discuss the differences in trends seen for measurements and calculations in the BWR case. This difference seems to indicate that neutron source strength in the BWR assemblies is increasing at a faster rate versus burnup than the calculations indicate. Because the neutron flux is mostly from ^{244}Cm, should there be zoning of enrichment levels among the fuel pins (with low enrichments to the outside), higher concentrations of ^{244}Cm to the outside would result. (In order to maintain uniform power, the neutron flux will increase in the low-enrichment areas, thus leading to increased actinide transmutation rates.) Thus, with enrichment zoning, neutron count rates could be higher than would be the case for assemblies with uniform enrichment and the neutron count rates would show a larger increase as a function of burnup.

Returning to the issue of ^{244}Cm underprediction, we note that the ORNL comparisons between experimentally determined concentrations and burnup simulation calculations involved both mixed-oxide (MOX) and low-enriched-uranium (LEU) fuels[7]. (Most of these investigations did not use the SCALE system, but some of the early investigations showed similar underpredictions using SCALE.) For both MOX and LEU, there was underprediction of ^{244}Cm concentrations relative to measurements. Two LEU samples were analyzed, and the results are relevant here. One LEU sample experienced a burnup of 31.2 GWd/t; for the other sample, the burnup was 52.4 GWd/t. Laboratory analysis and burnup-simulation calculations resulted in a calculated-to-experimental (C/E) ratio of 0.84 for ^{244}Cm in the first sample, with a measurement uncertainty of 3% (95% confidence level). The second sample was analyzed independently at two separate laboratories and yielded C/E ratios of 0.73 and 0.86. The experimental values for the second sample had uncertainties of 13 and 3%, respectively. On the basis of these results, it is reasonable to assume a C/E ratio of 0.85 for ^{244}Cm and to increase the neutron source-strength estimates accordingly. This would improve the agreement between calculations and measurement considerably.

10. Photon results

We have carried out photon (gamma) transport calculations for the 9 × 9 BWR assemblies; however, no gamma calculations were carried out for the PWR cases. Figure 12 shows the gamma results for the 9 × 9 BWR assemblies. The experimentally measured values are shown as well as the calculated values obtained by using ICRP flux-to-dose-rate conversion factors. The agreement between calculation and experiment is encouraging, although the predictions tend to be high. A later version of ICRP conversion factors exists and would likely lower the estimates by about 10%. The ICRP values that were used were readily available and were considered appropriately conservative for shielding studies. Again, in a practical application, a normalization factor is probably required.

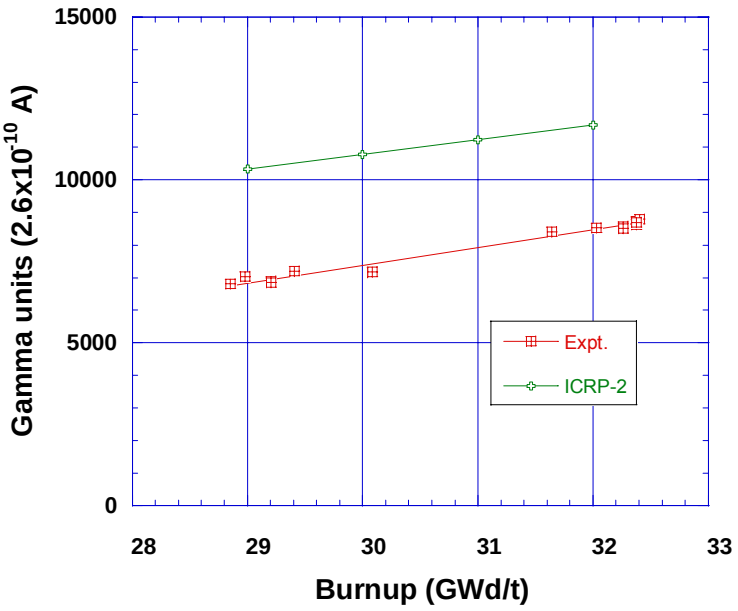

Figure 12. Gamma results for the BWR assemblies. Although the calculated values are higher, the trends are similar.

11. Summary and discussion

Monte Carlo models were developed that analyze Fork detector operation with both BWR and PWR spent-fuel assemblies. With spent-fuel characterizations obtained using the SCALE system, response functions were calculated for neutron and photon measurements and for both assembly arrangements.

Using monitoring data obtained by EURATOM, we performed benchmarking studies that compare trends between the measured data and model predictions. The results of the benchmarking studies are encouraging, but there are questions that remain to be answered.

The observed count rates in the neutron detectors are higher than what are predicted. Measured and calculated trends as a function of burnup are similar in the PWR case but they differ in the BWR case. A likely explanation for the higher observed count rates is that the neutron source strengths may have been underestimated. The different trends predicted for the BWR assembly may be the result of enrichment zoning.

Analysis of the photon response shows predictions to be about 50% above the measured values when using one set of flux-to-dose-rate conversion factors. One possible issue is the accuracy of the flux-to-dose-rate conversion factors. Allowing for uncertainties as to threshold settings and other possible calibration issues, we feel that the agreements are reasonable and trends seem to be as predicted.

In summary, this work gives us the ability to specify an arbitrary neutron or gamma source distributed within a spent-fuel assembly and then to estimate the measured signals in a Fork detector. In a practical application, normalization factors would probably be required. But, a number of factors affecting Fork detector measurements need further investigation. In particular, it would seem worthwhile to investigate the detector performance as a function of the boron content in the cooling pool. We also discussed the effect of enrichment zoning in fuel assemblies. If the outer rods in an assembly have enrichments that are different from those of the inner rods, this factor would quite likely significantly affect the Fork detector response. The models developed here together with the two-dimensional burnup-simulation model in SCALE are well suited to address the important issue of zoning in fuel assemblies. This consideration is quite important if MOX fuel assemblies are being monitored.

12. References

1. Rinard, P.M.; Boser, G.E.; *Safeguarding LWR Spent Fuel with the Fork Detector*, LA-11096-MS; Los Alamos National Laboratory, Los Alamos, NM; March 1988.

2. *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations*, ORNL/TM-2005/39, Version 5, Vols. I–III; Oak Ridge National Laboratory, Oak Ridge, TN; April 2005. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-725.

3. Briesmeister, J. F., ed.; *MCNP—A General Monte Carlo N-Particle Transport Code—Version 4C*, LA-13709-M; Los Alamos National Laboratory, Los Alamos, NM; April 2000.

4. Personal communication to B. D. Murphy, Oak Ridge National Laboratory, from EURATOM staff; February 2004.

5. ANS-6.1.1 Working Group, M. E. Battat (Chairman); *American National Standard Neutron and Gamma-Ray Flux-to-Dose Rate Factors,* ANSI/ANS-6.1.1-1977 (N666); American Nuclear Society, LaGrange Park, IL; 1977.

6. ICRP Committee 3 Task Group, P. Grande and M. C. O'Riordan (Chairmen); *Data for Protection against Ionizing Radiation from External Sources: Supplement to ICRP Publication 15*, ICRP-21; International Commission on Radiological Protection; Pergamon Press, Oxford; April 1971.

7. Murphy, B. D; Primm III, R. T.; "Simulation of Mixed-Oxide and Low-Enriched Uranium Fuel Burnup in a Pressurized Water Reactor and Validation against Destructive Analysis Results," *Nuclear Science and Engineering* **142,** 258-269 (2002).

MCNPX for Neutron Multiplicity Detector Simulation

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Abstract:
MCNPX is a multi-particle transport code that is widely used in the Safeguards community for the design of neutron detectors and the interpretation of measured results. Recent modifications to the code have improved its capability to model neutron coincidence counters and neutron multiplicity counters. The new features include:

- *Spontaneous fission sources*
- *Improved modelling of the multiplicity of induced fissions*
- *Tallying of coincidence capture events*
- *Creation of pulse output files*
- *Tallies that include the predelay and gate widths of recording electronics*

These additions allow the user to calculate all of the multiplicity counting rates in neutron detectors up to any moment of the distribution (Singles, Doubles, Triples, Quads….).
MCNPX now allows these rates to be calculated without the need for the assumptions of the "point model", thus allowing the calculation of multiplicity counting rates in arbitrary geometry. Because MCNPX is a general-purpose transport code, these new features can be used for all neutron counting cases, both passive and active and "good" geometry (well-counters) and "bad" (e.g. waste counters with multiple dieaway times). The yield and multiplicity distributions of the spontaneous and induced fission have default values that can be modified by the user.
These developments are a major contribution to the field of modelling of neutron multiplicity counters and will enable both more widespread and more accurate applications. The code has been tested against experiments and calculations of simple well-counter detectors and has shown excellent agreement.
The paper will describe the new features of the code and will show some results of test cases.

Keywords: MCNPX, Monte Carlo, neutron transport, coincidence counting, multiplicity counting, simulation

1. Introduction

The intention of this paper is to inform the safeguards community about the latest status of the MCNPX [1] code for the calculation of NDA instrument performance. MCNPX is a general purpose neutron and charged particle tracking code that operates over a wide energy region. Important improvements have been made to both the physics modelling and the tallying that allow the direct calculation of the counting rates of neutron coincidence (and multiplicity) detectors. Previously Monte Carlo codes were mostly used to calculate detector efficiency and dieaway time and sample multiplication in order to calculate counting rates from the point model equations. Both the MCNP-REN [2] and the MCNP-PTA [3] codes are modifications of MCNP that provide some capability for multiplicity calculations. However the current implementation is the first time these capabilities have been provided in a widely available, fully supported code. The results, including statistical uncertainties, are directly available in the code output without a need for post processing. The features of the code have been introduced in a number of versions. The latest (non-beta) version with all these features is 2.5.0 available from RSICC, the Radiation Safety Information Computational Centre in Oak Ridge, TN, http://www-rsicc.ornl.gov/.

2. New Features

This section describes the new features that are relevant to neutron multiplicity counting. It should be noted that the code also contains other new features that are described in [1].

2.1. Spontaneous Fission Sources

The general source was modified to allow the use of different spontaneous fission sources. Twenty-six default nuclides are available: ^{232}Th, ^{232}U, ^{233}U, ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{237}Np, ^{236}Pu, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am, ^{242}Cm, ^{244}Cm, ^{246}Cm, ^{248}Cm, ^{249}Bk, ^{246}Cf, ^{250}Cf, ^{252}Cf, ^{254}Cf , ^{257}Fm and ^{252}No. Additional nuclides may be added with a new input (FMULT) card.

The source is defined in the usual way (with the addition of par=sf). The spontaneous fission nuclide is selected from the nuclides defined in the source cell. If more than one spontaneous fission nuclide is in a source cell, the fissioning nuclide will be chosen proportionately to the product of its atom fraction and the spontaneous fission yield for each nuclide. If no spontaneous fission nuclide is found in a specified source cell, the code exits with a bad trouble error, “spontaneous fission impossible.”

The number of spontaneous fission neutrons is then sampled. The spontaneous fission multiplicity values have default values that can be modified. The energies are sampled from a Watt spectrum with appropriate spontaneous fission parameters for the selected nuclide. The code prints out the values used for all nuclides. Table 1 is an extract from this output.

1fission multiplicity data. print table 38

zaid	width	Watt1	Watt2	yield	sfnu									
90232	1.079	0.800	4.0	6.E-8	2.14									
92232	1.041	0.8548	4.0321	8.6e-4	1.760									
92233	1.079	0.89220	3.72278	1.3	1.710									
92234	1.079	0.77124	4.92449	5.02e-3	1.81									
92235	1.072	0.77471	4.85231	2.99e-4	1.86									
92236	1.079	0.73517	5.35746	5.49e-3	1.91									
92238	1.230	0.64832	6.81057	1.36e-2	0.048	0.297	0.722	0.95	0.993	1.0	1.0	1.0	1.0	1.0
92237	1.079	0.83344	4.24147	1.14e-4	2.05									
94236	0.0	0.0	0.0	0.0	0.08	0.293	0.670	0.905	0.980	1.0	1.0	1.0	1.0	1.0
94238	1.115	0.84783	4.16933	2590	0.056	0.267	0.647	0.869	0.974	1.0	1.0	1.0	1.0	1.0
94239	1.140	0.88525	3.80269	0.022	2.16									
94240	1.109	0.79493	4.68927	1020	0.063	0.295	0.628	0.881	0.980	0.998	1.0	1.0	1.0	1.0
94241	1.079	0.84247	4.15150	0.05	2.25									
92242	1.069	0.81915	4.36668	1720	0.068	0.297	0.631	0.879	0.979	0.997	1.0	1.0	1.0	1.0

Table 1: Source Multiplicity Data Output

The code also prints out the multiplicity and moments of the source used in the problem.

Another addition to the general source is the ability to use a mixture of source particles on the source card. This allows a spontaneous fission source to be combined with a single neutron source to represent the case of a sample with both spontaneous fission and (α,n) neutron emission.

2.2. Multiplicity of Induced Fissions

The modelling of multiplicity for induced fissions was also modified. Instead of simply sampling the integers either side of the average ν value, the number of neutrons is selected from a Gaussian distribution. The default FWHM of this distribution is 1.079. Some particular nuclides have specific values for the width of the distribution. These are shown in Table 2.

Nuclide	FWHM
92233	1.041
92235	1.072
92238	1.230
94236	1.110
94238	1.115
94239	1.140
94240	1.109
94242	1.069
96242	1.053
96244	1.036
98252	1.207

Table 2: Induced Multiplicity Widths for Selected Nuclides

The code prints out the multiplicity and moments of the source including induced fission.

The old multiplicity method can be selected for comparison purposes and backward compatability.

2.3. Improved S(α,β) Treatment

The S(α,β) thermal neutron treatment user interface is unchanged, but the underlying secondary energy treatment is improved. The S(α,β) thermal neutron treatment in many cases uses discrete energies instead of 32-equiprobable bin histogram energies. The result is non-physical spikes in the thermal neutron spectrum. These spikes are now smoothed out by a new algorithm.

2.4. Tallying of Coincidence Capture Events

A new option in tallying is to calculate a histogram of the number of captures (in an arbitrary combination of cells and nuclides) per history. From this histogram the (reduced) moments of the distribution are calculated. This can be done for an infinite gate length or for practical values of predelay and gate length. The calculation is done in an analogue way by counting all of the pulses (in the window) following each pulse. This means that currently all of the MCNPX runs that use this tally must be analogue. No variance reduction is allowed.

The moments resulting from these calculations allow Single, Doubles, Triples etc. rates to be calculated directly. (Previously, the Doubles and Triples gate fractions had to be determined separately. This was often done by Monte Carlo by calculating the time distribution of neutrons in the detector and fitting a number of exponential curves to it.)

2.5. Creation of Pulse Output Files

It is now possible to write a file that contains the position and time of each interesting capture event, where interesting is usually a capture that represents a detection of some kind. The files are written with an extension to the existing PTRAC file capability. These files can be used to create pulse train files that simulate a stream of pulses from a particular detector. The pulse train files can be analysed in software (external to MCNPX) that produces the Singles, Doubles and Triples using a “software shift register.” This data can be used to study the effects of deadtime and correction algorithms, which cannot currently be done inside MCNPX. These pulse trains are currently being used as part of the ESARDA NDA Working Group Multiplicity Benchmark Exercise.

3. Examples

These features have been developed over the past couple of years and a number of calculations, both passive and active, have been done to compare the results with both experiment and point model equations.

3.1. Epithermal Neutron Multiplicity Counter

An MCNP model of an Epithermal Neutron Multiplicity Counter (ENMC) [4] was used to calculate the Singles, Doubles and Triples counting rates to compare with measurement of a ^{252}Cf source. This was done for a range of gate widths from 5 to 40 μsec. The difference between the measured and calculated values for the Singles count-rate was 1.5%. This is reasonable, taking into account the uncertainty in the ^{252}Cf source strength. The Doubles and Triples have systematic differences of 2% and 3% respectively (Figure 1). (The uncertainty on each value is smaller than the size of the symbol). It may be noted that the agreement between measurements and calculation for a particular counter should not be expected to be better than the difference between two nominally identical counters.

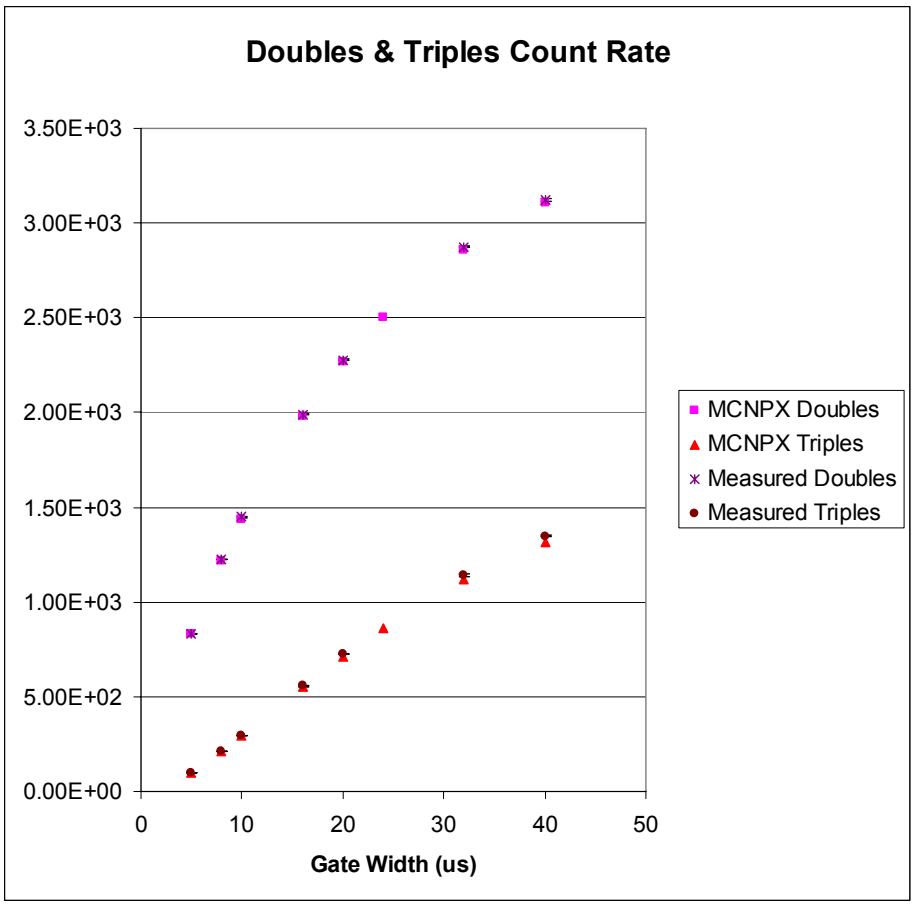

Figure 1: Doubles and Triples Counting Rates for the ENMC as a function of gate width for ^{252}Cf

The time behaviour of the neutron dieaway curve is fitted well by a sum of three exponentials, having approximately equal amplitude and time constants of 6.4, 14.2 and 29 μsec.

3.2. Sample Multiplication

The counting rates of a sample measured in a 5 ring multiplicity counter were compared to MCNPX calculations [5]. The model consisted of a large, 100% efficient counter in order to study only sample multiplication effects. The known counter efficiency and gate widths were used to match the calculated

values to the measured results. The difference between measured Doubles and calculated Doubles was 0.2% and for Triples the difference was 3%.

3.3. AmLi spectrum

Although this subject is not directly related to the new capabilities of MCNPX, the neutron spectrum of americium-lithium sources is important in many active neutron systems in safeguards. The calculations described in section 3.1 were continued using an AmLi source in the central cavity of the ENMC. The counting ratio of the 4th and 1st rings depends on the energy of the source neutrons and therefore a comparison between measured and calculated values should give an indication of the validity of the presumed spectrum. Four different formulations of the AmLi spectrum were used: a) a Russian measurement from Obninsk [6], b) "historic" values, c) the values of Geiger and Van der Zwan [7] and d) the values of Tagziria [8]. The spectra, normalised to the same areas are shown in figure 2.

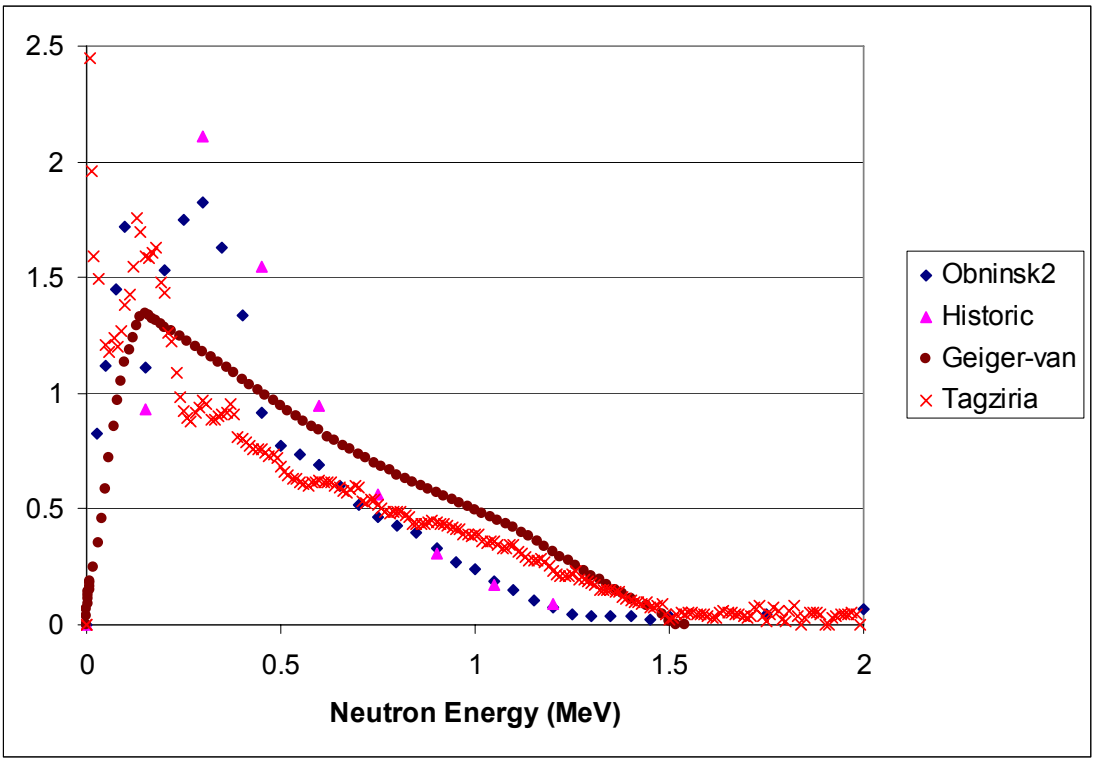

Figure 2: AmLi spectra

The results of the MCNPX calculations of ring ratio are plotted in figure 3, and given in Table 3 which shows in addition the ring ratios for Pu samples and a ^{252}Cf source. The horizontal line is the measured ring ratio for an AmLi source. The uncertainty in measured and calculated ratios is smaller than the size of the symbols. Despite significant differences in the spectra themselves, which could be due to the particular source manufacture and encapsulation, a) and d) give the closest calculated ring ratios to the measured one. It is not surprising that the Geiger spectrum, c), is harder and gives a higher ring ratio since it is the primary spectrum without source scattering effects, which were not modelled here. The use of the Tagziria data appears appropriate when the escaping neutron spectrum is desired.

Source	Mean Energy (MeV)	Ring4/Ring1
Measured:		
Pu	1.98	0.59
Cf	2.1	0.59
AmLi	0.49	0.26
Calculated:		
Pu	1.98	0.605
Cf	2.1	0.616
a) AmLi Obninsk	0.378	0.243
b) AmLi historic	0.492	0.260
c) AmLi Geiger et al	0.54	0.300
d) Tagziria	0.478	0.255

Table 3: Mean Energy and ring ratios for the ENMC

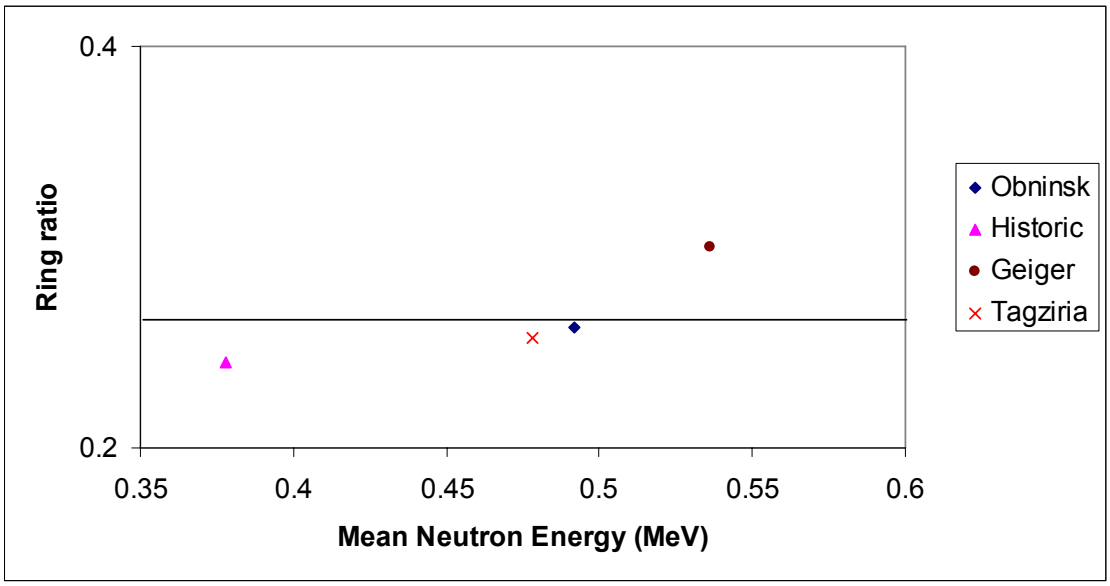

Figure 3: Calculated ring ratios for AmLi sources in the ENMC

4. Conclusions

MCNPX now allows realistic simulation of multiplicity detector counting rates. It is possible to calculate such rates even when the assumptions necessary for the point model do not apply. The results of a few comparisons give good agreement with experiment. This new capability will simplify multiplicity simulations of large items, such as fuel assemblies, that previously required the sample to be split up into many small pieces in order to satisfy the point model assumptions.

There are several areas of neutron multiplicity analysis that need further development. One is the effect of deadtime. The pulse trains produced from this code will allow investigations of this problem. The second are is the analysis of measurement data to produce a measured mass. Although we now have a convenient tool for predicting counting rates from sample parameters, in the majority of cases we still rely on the point model equations (with ad hoc corrections) for the inverse problem of producing sample parameters, primarily the mass, from counting rates.

5. References

1. J. Hendricks et al. “MCNPX Extensions Version 2.5.0” Los Alamos National Laboratory Report LA-UR-05-2675, http://mcnpx.lanl.gov/ (April 2005)

2. M. Abhold and M. Baker “MCNP–REN: a Monte Carlo tool for neutron detector design” Nuclear Instruments and Methods in Physics Research A 485 (2002) 576–584

3. M. Looman, P. Peerani, P. Schillebeeckx. “An Overview of NDA Instruments Modelled with the MCNP-PTA code at JRC Ispra”, Proceedings of the 23rd ESARDA Symposium on Safeguards & Nuclear Material Management, Bruges, Belgium, 8-10 May 2001

4. J.E. Stewart, H.O. Menlove, D.R. Mayo, W.H. Geist, L.A. Carrillo and G.D. Herrera “The Epithermal Neutron Multiplicity Counter Design and Performance Manual: More Rapid Plutonium and Uranium Inventory Verifications by Factors of 5–20” Los Alamos National Laboratory Report LA-13743-M (2000)

5. M. Krick Private Communication March 2005.

6. P. Rinard Private Communication 1998.

7. K Geiger and L Van der Zwan Health Physics 21 (1971) 120

8. H. Tagziria, N. Roberts, D. Thomas “Measurement of the 241Am-Li radionuclide source spectrum” Nucl Instr. And Methods in Physics Research A 510 (2003) 346-356

Session 8

Aspects of Non-Proliferation

Global methodologies & Technologies for nuclear non proliferation regime and other regimes.

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Abstract:

In the late eighties, the Military Application Division of the French Atomic Energy Commission (CEA) started to develop a research programme on nuclear proliferation to answer emerging threats in the context of the end of the cold war and the risks of nuclear weapons dissemination. This programme relies on in-house skills of the CEA (nuclear fuel cycle, nuclear weapons fabrication and advanced technologies) and on a network of academics, institutes and high-tech companies. This wide span range of expertises allows carrying in-depth analyses and assessments of all the steps and all the aspects of an undeclared nuclear weapon programme, the possible means of delivery and the technologies to monitor them. Researches on proliferation are carried out to the benefit of French authorities and to support international body activities (IAEA, CTBTO, INVO, UNMOVIC). Assessments of nuclear proliferation are achieved through the implementation of a methodology based on monitoring and verification technologies, some very sophisticated, remote detection and sensing, forensic detection, information analysis, at different stages, from the early detection of the willingness of a proliferation capable state, the building of a proliferation scenario, the detection and assessment of clandestine programme to the confirmation and the follow up of an established military programme. All kind of WMD proliferations follow more or less the same path and goes through the same sequences, requiring somehow the same type of monitoring and verification techniques. The multidisciplinary development of the CEA in the field of advanced sensors, information technologies and biosciences has allowed extending globally the approach of nuclear proliferation to the biological, ballistic and chemical proliferation. This work details this methodological approach and its extension to the other WMD proliferation. It tries to show the benefit, both politic and strategic to address globally the different types of proliferation, including the question of terrorist threats.

Keywords : global WMD proliferation approach, monitoring and verification methodologies and technologies;; monitoring technologies and sensors, synergies.

1. Introduction

The French Atomic Energy Commission has been established as the Technical Advisor of the Government and the Military Applications Division (DAM) of the French Atomic Energy Commission (CEA) is in charge of all technical questions related to nuclear weapons and the proliferation of nuclear weapons. The non proliferation activities carried out at DAM, aim to detect proliferation intentions of an alleged country as soon as possible, to assess the extend of a country's proliferating programme, to provide the French Authorities with analysis and technical evidences or at least clues of proliferation or non compliance to able them to establish a position and make up the case in ad hoc fora and finally support international organisations like the IAEA to help them to discharge their responsibilities.

To meet these goals, methodologies and technologies of monitoring and verification were developed along more than 20 years to help stemming nuclear proliferation. In the late nineties, in the perspective of

the growing threat of biological weapons, the discovery of the bio weapons programme of the former USSR and the increasing threats of WMD's means of delivery, the importance of a coherent approach of state proliferation as a whole was pointed out. It appeared that the all WMD proliferations could, to some extend, be addressed using common methodologies and close technologies. On one hand, one could note that most states of concern, seeking to develop a nuclear weapons programme, are also developing theirs means of delivery and often, the nuclear programme comes after or comes with bio/chemical weapons researches and developments. On the other hand, many tools developed for the monitoring and verification of nuclear non proliferation could somehow be applied to the monitoring and surveillance of other WMD proliferation. The meaning of this remark is, that addressing WMD proliferation globally not only brings cost saving in R & D but also improves the efficiency of the fight against proliferation and terrorism. In consequence, synergies and cross cutting R & D between teams, laboratories, academics and multinational organisations must be encouraged. This paper describes how proliferation could be addressed globally and proposes some foods for thoughts on the recent crises of the nuclear non proliferation regime.

2. An historical overview.

2.1. Involvement of the CEA in non-proliferation activities.

The French Atomic Energy Commission has been involved in non proliferation since a long time. Having responsibilities of civil nuclear fuel cycle R & D and of nuclear weapon fabrication, it has gained all the knowledge and expertise necessary to deal with all aspects of non-proliferation related issues. In 1958, it has been established as the Technical Advisor of the Government by a Prime Minister order for all issues related to the atomic energy and is the government representative to International Organisation in charge of nuclear questions (e.g. IAEA).

2.1.1 Organisation.

Inside the CEA, two Divisions are dealing with non-proliferation matters. The Division for International Affairs (DRI) has the responsibility for the nuclear policy management, especially in the area of nuclear non proliferation and export control of sensitive or dual use nuclear technologies and under the authority of the Ministry of Foreign Affairs, relation with international organisations like IAEA, The Military Applications Division (DAM) of the French Atomic Energy Commission (CEA) is in charge of all technical questions regarding nuclear weapons, the non proliferation of nuclear weapons and related verification and monitoring issues.

Since the late eighties, the Military Application Division (DAM) of the Atomic Energy Commission (CEA) has developed a coherent non proliferation research and development programme to give French Authorities the technical means necessary to prevent and answer the growing threats of nuclear proliferation and terrorism. This programme which includes monitoring and surveillance methodologies and technologies, relies on in-house skills of the CEA (nuclear fuel cycle, nuclear weapons fabrication and advanced technologies) and on a network of academics, institutes and associated high-tech companies.

2.1.2 Adaptation to a changing world

In the early nineties, the discovery of the clandestine nuclear weapon fabrication programme of Iraq in the aftermaths of the Gulf war and the finding of North Korea's undeclared nuclear reprocessing activities, get the CEA to provide technical assessments to the French Authorities and to support the verification activities of IAEA/Action TEAM and UNSCOM for the implementation of UNSC Resolutions and the establishment OMV plan to disarm Iraq and help the IAEA, to assess the extend of DPRK undeclared nuclear activities and implement the "freeze" agreement in this country.

During these years, the efforts on Iraqi nuclear programme related issues of verification and monitoring gave a shot to non-proliferation R & D in various areas as: countries of concern proliferation programme assessment, analysis of nuclear fuel cycle, nuclear weapons, programme, monitoring and verification

methodology, definition of observables and clues and the technologies to detect them, feasibility of technologies implementation, etc...

In 1997, the research and developments projects carried out on nuclear non proliferation and nuclear and radiological terrorism issues have been positioned under the umbrella of the overall joint task agreement between the CEA and the Ministry of Defence.

2.1.3. New threats

In the late nineties and the early 2000's, to tackle the emergence of the threats of biological weapon proliferation and new means of delivery and their possible use by non state actors, the field of interest was extended to chemical, biological and missile proliferation. In 2001, the 9/11 terrorist attacks and its following has speed the work on the prevention of terrorism acts and mitigation of their consequences. Since that time, important efforts are going on to further develop and qualify new detection methods.

2.2. Value of the Iraqi and North Korean experience.

The years after, the Gulf war and the disclosure of the Iraq's clandestine WMD activities, the disarmament of Iraq under the UNSC Resolutions gave a shot to the non proliferation programme. The support given to the IAEA and UNSCOM to help them to discharge their responsibilities in destroying or rendering harmless all the significant components of the Iraqi WMD related equipments and facilities and to establish a credible Ongoing Monitoring and Verification Plan (OMV Plan) to prevent Iraq from resuming prohibited activities were strong incentives to develop a robust non proliferation programme. At the same time, came the North Korean crisis and the discovery of its undeclared nuclear activities in violation of its safeguards agreements along with the suspicion of weapon grade plutonium production. The support to the AIEA to assess the extension of the DPRK undeclared activities confirmed the objectives of the non-proliferation programme set up in the Iraqi framework. Now, some years after, it can objectively be said that as for many other organisations, the contribution to the elimination of WMD and the establishment of the monitoring in Iraq, was a real "test-bed". Iraq from 1992 to 1998 was a country-wide, open field laboratory where many advanced monitoring technologies were experimented, allowing supporting states and organisations to test new monitoring approaches, new verification concepts and new technologies as well as to conclude new legal instruments, all valuable to deal with to day's crisis (e.g. the additional protocol, the dual use items export control, environmental sampling, use of satellite imagery, open sources analysis, etc..).

2.3. Extension to other WMD proliferations.

In the late nineties, from the lesson of past experiences and appearance of new threats, came out the awareness that different types of proliferation follow more or less the same path. Over some time, the approach of WMD access by a State is a global one. If the State in seeking to acquire chemical weapons, it may try to get biological weapons and if its technological level allows then, it will decide to get nuclear. The idea that a more global approach of WMD proliferation as a whole is needed has then taken shape. Such an approach was achieved by drawing from the experience gained in the nineties and taking advantage of the large multidisciplinary range of activities in the CEA.

- Nuclear fuel cycle, nuclear material and related activities;
- Bio and information technologies;
- Modern technologies, advanced material and process;
- Fundamental and applied physic;

This set of in-house skills and expertises is supported by a network of institutes and high-tech companies. Then, through cooperation between all these bodies, the CEA is extending its domain of R & D in non proliferation activities to the chemical and biological proliferation. With these developments the CEA was able to also address terrorism threats.

The goal of theses new developments is to answer to the emerging threat of biological and chemical proliferation and terrorism through the timely detection of prohibited activities by giving the tools to detect

and assess the proliferation in a country at is earliest to provide the French Authorities with the right information to make up their own assessment of the situation.

Furthermore, theses developments come at a time the EU members willing to address the threats of WMD and terrorism and to build security in European Union neighbourhood by strengthening the European Security and Defence Policy. EU members have adopted a Common Security Strategy against proliferation of weapons of mass destruction and Terrorism [1]. In that context, a Preparatory Action on Security Research (PARS) [2]has been launched to enhance the European industrial potential in the field of security and prepare a large programme to improve European security through Research and Technology. The CEA and the DAM is taking part of PARS in relation with national and european industrial companies and research institutes.

3. NBC & M Proliferation development scheme and monitoring

Dealing with the specific problems of Iraq and North Korea, the first objective was to ascertain the extend of the proliferation programme and to describe all the steps of the nuclear weapons manufacture cycle, **“from the mine to the weapon”** from "known (verified) information and see what was the level of uncertainties brought by unknown or unverified information. This work included studies, to reconstruct the programme from the known + assumptions on the unknown. This method works also for the new crisis which occurred since. Then, from the identification of critical points in proliferating programme, theoretical and experimental studies were carried out, the purpose of which was to define observable parameters and clues for and to develop the instruments able to detect them.

3.1. State proliferation programme.

The path that followed a proliferation programme is more or less always the same, whatever the kind of weapons of mass destruction or means of delivery it aims to acquire. It will go trough the same steps and the same milestone. The incentives for the political and strategic decision to acquire WMD are diverse: a certain international or regional security context, a quest for hegemony, etc. Once decision is made the State has to set up an organisation, mainly covert (cf. the PC3 Iraqi organisation) which organise the interdisciplinary work of the programme along all the steps (fig.1) from scratch or from the level of knowledge already openly acquired, through R & D at laboratory scale, pilot scale experiment research and industrial phases with the building of production facilities, the weaponization and testing facilities and finally, the stockpiling with most of the work realised in clandestine (undeclared) facilities or in misused (declared) facilities.

At each step, the State may rely on clandestine procurement and transfer networks for acquiring the necessary scientific and technical knowledge, known how, equipment, material and technologies and train staff abroad as did the one headed the scientific Pakistani by A.Q. KHAN and the A.Q. KHAN Laboratories to help Libya, North Korea and probably other countries. Transfer through a proliferation network has already been done on exchange basis eg balistic technology in return of nuclear one. Assistance and support through a proliferation network could bring critical help putting forward the completion of the programme. In consequence, such scenario should be included in the analysis of the problem.

3.2. Proliferation development phases.

Figures 2 describe the phases of any proliferation programme. Once the decision has been taken, the first step will be to put in place a cover interdisciplinary organisation, define a programme, secure the funding, recruit specialists in the different areas of material production, fabrication and weapons development, possibly abroad, establish intelligence networks to collect information and know-how, send experts abroad to participate to key seminars and symposiums. Then the R & D could start at laboratory scale, then pilot scale and the building of production facilities. At the end, comes the weaponization with the need for specific facilities as test field, assembly buildings or filling facilities.

As seen earlier, at any steps, the needs in know-how, raw materials, laboratory equipment, machine-tools, dual-use equipment, could be fulfilled either indigenously or abroad through procurements and recruitments networks. The whole programme could stretches over tens of years, depending of the scientific and industrial development of the country.

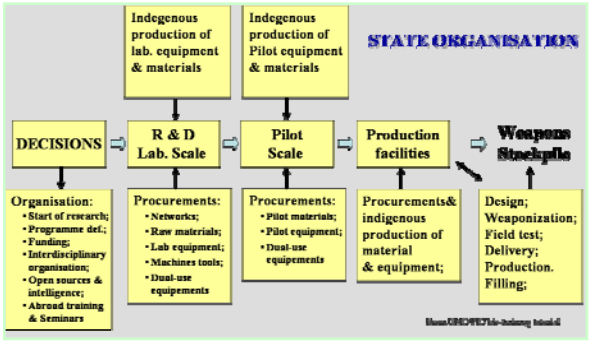

Fig.: 1 NBC & M proliferation programme scheme

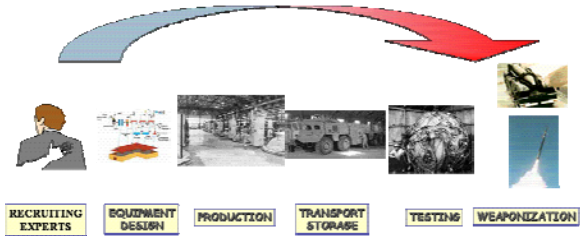

Fig.2a: Proliferation & Terrorism development phases schematisation.

3.3 Proliferation monitoring and verification strategy

From the description of the different phases of proliferation path (including external procurement & assistance), indicators (clues) and observables are defined and modelized which will be the basis for the development of a proliferation monitoring strategy. Cross-cutting issues and observables of the same kind for different type of proliferation (e.g. satellite imagery) allow developing synergies, the benefit of which result in improved efficiency, sensors sharing and resources saving.

On that basis a monitoring and detection strategy is elaborated which means the development of monitoring systems/schemes and the definition, elaboration and implementation of suitable tools, e.g. procurement networks analysis, process observables definition to define parameters for the remote monitoring: spatial imagery, geophysical, radiological survey, environmental monitoring, chemical and radioactive species detection and so on…

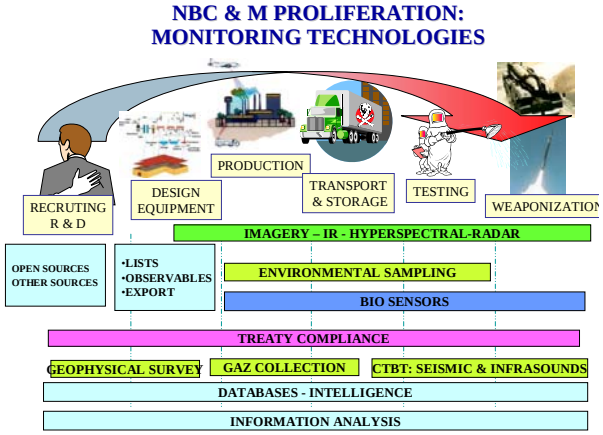

Fig.2b: Proliferation Monitoring techno phases schematisation.

The most important goal of the strategy is to allow a timely detection and assessment of intention as early as possible or if not, to reach a high level of assurance of the detection before the WMD programme

comes to its full, reaching a point where compelling the state to stop and go backward becomes much more difficult. A timely detection and a reliable analysis of the situation allow providing French Authorities and International organisation (e.g. IAEA) with relevant information, in view to trigger a political action. In that perspective, monitoring techniques like experts and Scientifics network follow-up, analysis of procurement attempt from export control data, satellite imagery, environmental sampling strategy, etc, should be considered and if needed, implemented by organisation in charge of the international law enforcement and supporting states.

The scenario of proliferation development had to direct the answers to the critical questions on assessment issues:

- Where to look? (e.g. facilities, transport, procurement);
- What to look for? (e.g. observables, clues);
- When to look? (e.g. at what stage of the process: training, fabrication, of procurement,…);
- How to look? (what sensors)
- What does the information means? (signal & information analysis/assessment)

From the outcomes of those questions, precedes the definition of a monitoring technologies development strategy.

3.4. Technologies supporting the monitoring and verification strategy

To answer the question of section 3.3 on where, what and how to look for, the CEA has developed a tool-box comprising a set of various technologies to be implemented to support action against proliferation. The following section give a look on some of them under development. In particular considering how to address new critical verification problems raised along the ten last years like as for the detection & verification of a bio and chemical programme (cf. the Iraqi programmes), a clandestine enrichment or reprocessing programme (Iran, Libya, North Korea), the identification of the actual destination of suspicious facilities, etc…. there is an urgent need for specific sensors and methodologies. The following section shows some features in satellite imagery, remote monitoring, environmental sampling, bio detection…

3.4.1. Satellite imagery (fig.3).

Satellite imagery is a good example of a remote monitoring technology. Satellite imagery monitoring is based on images from commercial or national satellite (National Technical Means) more often a combination of both and the use of different types of satellite images, panchromatic, infrared, hyper-spectral and radar, along with advanced image processing and data fusion. Satellite imagery is very promising for:

- Detection of undeclared (clandestine) activities.
- Deterrence: to deter any covert WMD programme, in particular nuclear weapon programme by increasing the programme cost, its duration and its risk of detection
- Optimization of the organisation of routine inspection or Complementary Access, challenge or special inspection.
- Enhance confidence

Satellite imagery allows some surveillance of a country of concern even if no inspection are carried out in the country (North Korea since 2002, Iraq 1998-2002) but it's not a surrogate. In some specific cases (Iraq) or if the verification agreement allows it (CFE, Open Sky treaty) aerial images from airborne sensors (planes, helicopters, drones) could be use in combination with the satellite images. New techniques are currently in development which would allow in the future to show up print of specific materials.

3.4.3. CTBT verification and its outcomes

The CTBT International Monitoring System (IMS) + the International Data Centre (IDC) + the National Data Centres (NDCs) supporting the IDC is also a good example of remote monitoring network. The researches and developments made for the implementation of the 321 stations of four technologies, seismic, hydroacoustic, infrasound, radioactive particle and radioactive noble gas (Xenon) air sampling, of

France is contributing to the realisation and operation of 25 stations of the four technologies and support actively the task of the Provisional Technical Secretariat to achieve the completion of the IMS network. (fig 4).

The International Monitoring System or for the CTBT On-Site Inspection have triggered significant progresses in other areas of the verification as for the safeguards implementations:

- Research on the detection of undeclared reprocessing activities through the ^{85}Krypton, a radioactive noble gas, detection and backtracking or detection of stable Xenon isotope characteristics of spent fuel storage are conducted based on experience gained on radioactive Xenon detection.
- Backtracking methods developed for the determination of the CTBT radionuclide network are also useful for the localisation of the emission source of other chemical species.
- Development of unattended sensor Integrating all functions: collection and processing, data processing, authentication and communication in a is also an outcome from the IMS researches (fig.5).

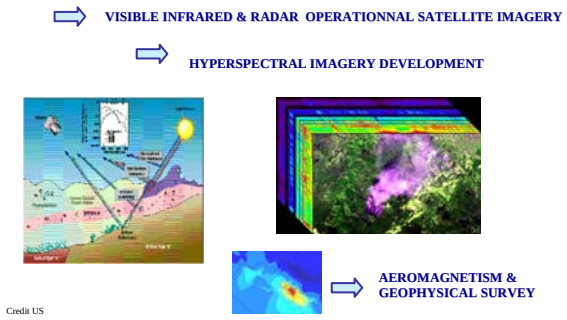

Fig.3 Remote detection / satellite/airborne imagery.
Credit of US DOE

The 25 stations of the french contribution to the CTBT IMS

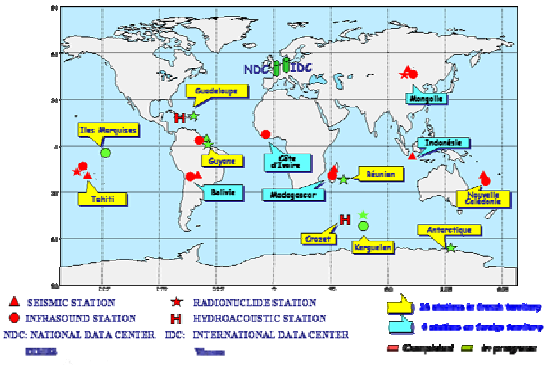

Fig 4: **Nuclear weapon test monitoring**
Credit CEA/DAM/DASE

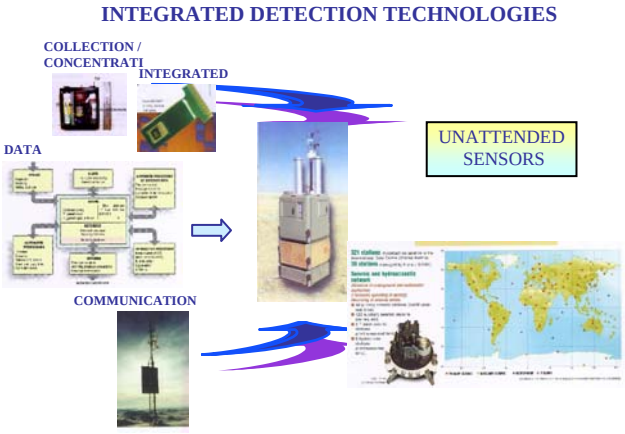

Fig.5: Example of Integrated detection technologies for CTBT IMS
Credit CEA/DAM/DASE

3.4.3. Environmental monitoring technologies

Environmental sampling and particle analysis is a very sensitive forensic technique which able to identify faint traces of fissile materials on swipes (fig 6a, b & c). This technique is implanted in the framework of the IAEA Support Programme and NWAL. It is the technique of choice to reveal undeclared activities or if applied in the framework of wide-area environmental sampling to detect clandestine activities of the nuclear fuel cycle. But it is also a very intrusive technique which request important means as a clean room for the sample preparation and a reactor for the irradiation of the preparation. It takes time,it is costly and could be implemented by only few laboratories which limits its use and should only used in a framework of well defined verification strategy.

Figure 6a, Microscope in clean room, class 10

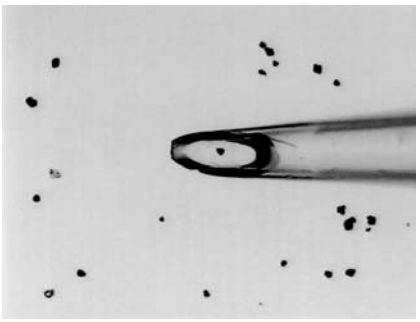

Figure 6b particle aspiration in a capilliary tube.

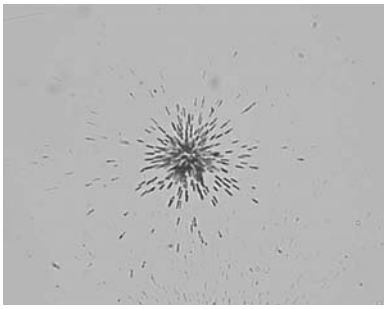

Figure 6c fission tracks in Lexan from a Unat particle.

3.4.5. Geophysical survey

Geophysical survey techniques (fig 7a & 7b) are a field of the verification which encompass a large span of techniques: microgravity, seismic probing, electromagnetism, magnetism, ground conductivity, ground penetrating radar which all request to be fielded by experts.. All methods aims to the detection of buried structures, underground undeclared facilities, tunnels, pipes, cavity. Gravimetric or electromagnetic methods could eventually be fielded airborne (drone). It has a large range of application from the CTBT/OSI to the implementation of IAEA safeguards. As for the environmental sampling, the Agency has laid down this need in its R & D programme for nuclear verification for example for Design Information Verification (DIV)

Figure 7a
Microgravimetry on fieldy

Figure 7b ground penetration radar / CGG
Credit CEA/DAM/DASE**.**

3.4.6. Information technologies

Information analysis is a very important tools to detect proliferation intention at it's earliest or along the life of the programme to detect undeclared research or fabrication facilities. Information could be issued from different sources, e.g.:

- Open sources: publication, scientific literature, news, international organisation reports, pics, and so on;
- Classified sources: national intelligence or bilateral/multilateral exchange (Nato, EU,…)
- Exchange of export denial on sensitive items;

Information analysis require the development of specific tools as for language translation, research of pertinent information, data fusion, presentation of information…

3.4.7. Bio and chemical monitoring

B & C proliferation monitoring require the development of specific sensors (e.g. lab on chips) the development of which benefits from the researches in other areas.

4. Role of legal instruments

Since the fifties, a number of legal instruments has been adopted by the international community to stop the spread of weapons of mass destruction and their delivery means. These legal instruments establish a non proliferation regime by setting rules to which states parties have to comply and for some them but not all, verifying compliance of undertakings through a protocol. As a framework, they are an important element of the global non proliferation monitoring.

4.1. Monitoring proliferation through compliance to legal instruments

Even if the efficiency of the following treaties and agreements could be questioned, considering their weakness, loopholes and lack of universal adherence, because they are the result of delicate compromise between negotiating parties with contradictory political goals, between efficiency & intrusivity and sovereignty & national security, legal international or multinational instruments are the first line of defence against the proliferation of weapons of masse destruction and their means of delivery (for example, in that perspective,, the Proliferation Resistance and Physical Protection working group of GEN IV International Forum and the INPRO group of the IAEA explore the value of legal instruments as "institutional barriers"). The policy of a country with respect to the non-proliferation legal instruments and its degree of cooperation with verification bodies are, therefore, a strong indicator of its desire to proliferate or to comply with international law. In particular, the denial of adherence to a set of legal instruments; a clear breach to its undertakings or a recurrent non cooperative attitude vis-à-vis of verification process, raise suspicion and provide clues on proliferation intentions and should direct more attention to that state (In that line the IAEA in the framework of the strengthening of the safeguards have set up an internal process of State Evaluation which allows to redirect resources where the actual safeguards concerns are).

As such, monitoring compliance with legal instruments is one important tool of the monitoring and verification strategy tool boxes. Let's have a brief review of the most important instrument.

4.2. Nuclear non proliferation regime.

The set of instruments dedicated to the prevention of the proliferation of nuclear weapons is the more comprehensive one with the Non Proliferation Treaty (NPT) [3] as a cornerstone and its verification body, the IAEA [4] Safeguards strengthened by the Additional Protocol [5], the control of exportation of basic nuclear items and dual-use items as described in the list of the Nuclear Supplier Group (NSG) [6], the prohibition of nuclear testing with the Comprehensive Test Ban Treaty (CTBT) [7] and in the future, a possible Cut-off treaty to ban the production of fissile materials for nuclear weapons. This set of tools which control the dissemination of nuclear weapons and related materials, fabrication technologies,

material production equipment and technologies, test of the weapons could seem to be sufficient to prevent, detect and stop any country embarked in a nuclear weapon fabrication programme. Nevertheless, recent crisis, Iran, North Korea, Libya, Pakistan and the unveiling of procurement and fabrication network has demonstrated a need for a strengthening of the control.

4.3. Biological & Chemical weapon prohibition.

The Chemical Weapons Convention (CWC) [8] and the Biological Weapons Convention (BWC) associated with the control of export of prohibited or dual use items set up in the Australia Group provide a framework to prevent the Chemical and the Biological weapons proliferation, but ones can point out that if the chemical weapons proliferation is monitored by the OPCW, the Bio Convention is still lacking of a verification tool and that the implementation of the Chemical Weapons Convention is not as it should be.

4.4 Ballistic & means of delivery: stemming the development and the spread

. The proliferation of ballistic missiles and other means of delivery as cruise missiles, drones or unmanned aircraft, is became a real concern. Missile development is strongly linked to the other types of proliferation as means of delivery and as items to exchange. to get in return knowledge and technologies in a network of proliferating countries, Pakistan, North Korea, Iran and others. The instruments put in place to deal with that issue, the Missile Technology Control Regime (MTCR) and the Code of Conduct, although first steps towards a better control, are insufficient to stop the development and the trade of means of delivery.

4.5. New tools for a new environment

In the first years of the new century a several events, some tragic like the 9/11 terrorist attacks, some other, North Korea, Iran, Libya, A.Q. Khan sensitive technologies black market, revealing loop holes in the prevention, all questioning the international security, brought the international community to strengthen the legal instruments or to forge new ones to prevent dangerous material and sensitive technologies to fall in bad hands. Then the Proliferation security initiative PSI to prevent trade of WMD and WMD components, UNSCR resolution 1540 to ban proliferation and impose States to adopt national law banning proliferation of WMDs, The G8 countries have launched the Global Partnership to secure fissile materials facilities and scientifics in the former USSR countries, the Global Threat Reduction Initiative which aims to secure radioactive material have been set up. As for the nuclear non proliferation regime, a proposal for a Multinational Nuclear Approach (MNA) of the nuclear fuel which aims to prevent the dissemination of enrichment, reprocessing and spent fuel storage, the most sensitive steps of nuclear fuel cycle, has been set up. In the future, negotiation for a Fissile material cut-off treaty may begin at the CD and the outcomes of the 2005 NPT review conference may add some new trends.

To be efficient, the implementation of all these new instruments will need to be supported by monitoring and verification by states parties or international bodies with the support of states parties. That's mean that the global monitoring and verification strategy should take them into account and be adapted as such.

5. SUMMARY

Development of Nuclear, Chemical, Biological and Missile proliferations follows more or less, the same path, from the decision by a government to acquire weapons of mass destruction, preparation, R & D activities, pilot scale production and tests, production facilities and constitution of stockpile with the possibility of foreign assistance at any stage and the procurement network. The methodologies applied and technologies implemented to detect nuclear proliferation at an early stage and to assess proliferation level of a state work also for Chemical, Biological and Missile proliferation. In that perspective, it appears that some the methodologies and the technologies developed to detect signatures and observable of a type of WMD may apply for other proliferations or to all type of proliferation. Synergies between R & D teams, companies and organisations must be encouraged to tackle the threat of proliferation as a whole.

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7. Disclaimer

The views expressed in this article and the presentation which goes with are only author's view and does not necessarily correspond to those of the French government or the French atomic Energy Commission (CEA).

8. References

[1] European Security Strategy; 12 December 2003; *A secure Europe in a better world*;

[2] Commission of the european communities; *On the implementation of the Preparatory Action on the enhancement of the European.Industrial potential in the field of Security research,Towards a programme to advance European security through Research and Technology,* 3.2.2004, COM(2004) 72 final.

[3] IAEA/INFCIRC/140

[4] IAEA/INFCIRC/153 (Corrected)

[5] IAEA/INFCIRC/540 (Corrected)

[6] IAEA/INFCIRC/254 Part. Rev.6 and Part. 2 Rev.5

[7] Preparatory Commission of the Comprehensive Test Ban Treaty; www.ctbto.org.

[8] Organisation for the Prohibition of Chemical Weapons (OPCW); www.opcw.org.

[9] IAEA/INFCIRC/640

Biosecurity Material Control and Accountability: Lessons for Nuclear Safeguards?

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Abstract:

Nuclear material safeguards have evolved over many years as a system to assure the non-proliferation of nuclear materials. To provide this assurance, both domestic systems of material control and accountancy (MC&A) and international safeguards employ accountancy as one key component.

Similar non-proliferation concerns have arisen more recently with respect to biological materials. By biological materials, we mean specifically those high-risk pathogens and toxins that may be considered potential sources for biological weapons or other malicious use. In attempting to devise appropriate measures for protecting high-risk pathogens and toxins against theft, or "laboratory biosecurity," we have investigated the relevance of experience from nuclear material safeguards. However, biological and nuclear materials have fundamental differences that demand modifications and even new approaches for making similar non-proliferation assurances.

The major difficulty in adapting nuclear approaches to biological materials is the concept of significant quantity. There is no such meaningful threshold for a replicating pathogen. The immediate consequence is the futility of material balance accountancy for bulk biological materials. Instead, we are forced to recast MC&A as "material control and accountability" in which the bookkeeping methods of accountancy are replaced by more qualitative methods of accountability, which enforce ownership and responsible stewardship of materials by accountable individuals.

Lessons from MC&A applied to biological materials could prove useful for application to nuclear safeguards. The International Atomic Energy Agency (IAEA) has employed detailed material-balance accounting to assure the non-diversion of nuclear materials; although successful, it may not be the best allocation of resources. Detailed accountancy is resource-intensive, competing with IAEA Additional Protocol measures of arguably increased importance from a broader nuclear non-proliferation perspective. Accountability measures developed for laboratory biosecurity may be accomplished efficiently yet effectively, and are therefore worth considering for nuclear material safeguards for indirect use materials. Improved safeguards efficiency for nuclear materials would facilitate the move toward Integrated Safeguards.

Keywords: safeguards; materials; accountancy; accountability; biosecurity

1. Introduction

The control and accountancy of nuclear materials have been essential components of the overall nuclear safeguards regime to stem the global proliferation of nuclear weapons. Other non-proliferation measures complement material control and accountancy by protecting nuclear materials and facilities, controlling the release of weapons-relevant information, restricting testing, and preventing the export of critical equipment.

A similar non-proliferation challenge has emerged more recently; namely, the safeguarding of biological materials that could be used as biological weapons, which is an essential component of "laboratory biosecurity." The general methodology and specific measures for laboratory biosecurity have been

under rapid development over the past few years, gaining added urgency following the anthrax attacks in the United States in October 2001. Because biological materials present unique challenges, biosecurity is not a simple matter of applying identical approaches developed for the nuclear case.

It may be tempting to imagine that simply destroying stocks of dangerous pathogens and toxins would be the most expedient solution for dealing with the biological threat. Indeed, such recommendations have actually been proposed, such as eliminating the known remaining stocks of the smallpox virus. However, such action may actually be counterproductive, because it cripples one's ability to develop defensive measures (e.g., vaccines, therapies) if the disease were to reappear. In many cases there are legitimate dual-use applications for the materials, such as treating muscle spasm conditions with botulinum toxin. Research on and diagnostics for infectious diseases would not be possible without maintaining stocks of dangerous materials, so there is really no practical way to avoid the issue. Storing and working with dangerous biological materials must continue.

In this paper we describe the material control and accountability aspect of laboratory biosecurity (biosecurity MC&A). Such a review is instructive to the nuclear safeguards community, in that it offers a perspective from which we can re-examine and reevaluate the nuclear case. Are there lessons to be learned that can help us improve nuclear safeguards?

2. Biosecurity

The threat from dangerous biological materials (pathogens and toxins, or "agents") could range from the diversion of pathogen stock material to help develop state-sponsored biological weapons, to the theft and intentional release of biological agents by terrorists. The malicious use of dangerous biological materials by disgruntled insiders or the destruction of critical vaccine stockpiles or unique culture collections are examples of other conceivable threats. While sources of pathogens and toxins can be found in nature, that is no excuse for permitting laboratory stocks to be left unprotected. Identified, well-characterized, isolated stocks of dangerous biological materials in laboratories need responsible stewardship that is both safe and secure.

"Laboratory biosecurity" is a collection of measures to reduce the risk of malicious use of biological materials.[1] Whereas safety has been a concern for many years and measures have been developed to prevent the accidental release of or exposure to pathogens, the necessity for security has been promoted only recently. The bioscience community now recognizes and accepts the need to prevent the deliberate, malicious use of dangerous pathogens and toxins.[2] While biosafety measures protect people from dangerous biological agents, laboratory biosecurity measures essentially protect biological agents from dangerous people.

Laboratory biosecurity is comprised of several components. Physical security of the facilities where biological agents are stored and worked on keeps outsiders away from the agents. Access controls allow only authorized individuals to enter facilities with biological agents. Personnel management ensures that only trusted individuals obtain such authorization. Special measures for chain of custody and packaging ensure the secure transfer of materials between laboratories. Information that would be useful to potential adversaries, such as the exact locations of materials and details of their associated security measures, is protected.

None of these components of biosecurity is sufficient by itself, nor is any given component perfect. A robust laboratory biosecurity approach therefore demands an additional element: the control of and accountability for the materials themselves. Over several decades in the nuclear safeguards community, we have developed and employed control and accountancy measures for nuclear materials. Much of that experience has been instructive in devising corresponding measures for biological materials.

3. Biological vs Nuclear Materials

In this section we summarize some of the main similarities and differences between biological and nuclear materials in considering how best to safeguard them.

3.1. Similarities

Both biological and nuclear materials are potential ingredients for weapons, and are therefore potentially attractive to both state proliferators and terrorists. The nature of these threats is also similar. For example, the insider

threat is common to both. Theft and sabotage are potential threats in either case.

Both biological and nuclear materials have legitimate, beneficial applications. They both involve sophisticated technology, knowledge, and infrastructure. They are potentially hazardous to those who work with them (e.g., infection from exposure to biological materials or radiation exposure and criticality accidents from nuclear materials) and thus require associated safety practices.

Biological and nuclear materials can both be created and destroyed through processes that exist in nature. Pathogens, whether they are bacteria, viruses, or rickettsiae, can replicate and die. Fissile uranium and plutonium can be created through neutron capture by fertile materials. Nuclear materials "die" through radioactive decay or fission.

Both biological and nuclear materials often exist as components of other things. Biological materials can exist in various animal, insect, or plant hosts, waste streams, or as environmental contamination. Nuclear materials are often alloyed, dispersed in ceramics or oxides, or deposited on substrate materials.

Both nuclear and biological materials have to be able to move between locations for a variety of reasons. For instance, samples of biological materials are transported to key reference laboratories for confirming diagnosis.

3.2. Differences

There are few nuclear material types of concern: uranium-233, uranium-235, and plutonium (except plutonium-238) are the principal "direct-use" nuclear materials for weapons. The number of different types of biological materials that could be described as "direct-use" is much greater. For example, the Centers for Disease Control and Prevention list 80 so-called "Select Agents" that are considered the most serious pathogens and toxins for humans, animals, and plants.[3] Although attempts are made to prioritize or reduce this list to more manageable proportions, the fact remains that there will always be many more variations of biological material types than nuclear.

Sources for biological materials are more widespread, even endemic. When natural disease outbreaks occur, they effectively become sources of materials, publicized in the open news media. Fortunately, safety considerations have enforced some consolidation, so that dangerous biological materials are usually confined to the limited number of laboratories designed for safe containment, namely those rated Biosafety Level 2 (BSL-2) and higher.

Biological materials can mutate unpredictably. They can even be "engineered." Such versatile manipulation is not an issue for nuclear materials.

Although environmental conditions affect biological materials, nuclear materials are not affected. Biological materials are often fragile. Many samples must be kept at low temperature; multiple freeze and thaw cycles can cause problems. Inventory processes can therefore damage the materials.

Nuclear materials have radiation signatures, but there is no such analog for biological materials. Radiation signatures enable nuclear materials to be detected indirectly, by analyzing the nature of the penetrating radiation they emit. This can often be accomplished on location with portable instrumentation. Except in rare cases, biological agents must be detected directly. Detection is not always possible in the field; it may require transporting a sample to a distant laboratory, culturing, isolating, and finally diagnostic analysis.

The concept of "significant quantity" does not exist for self-replicating biological materials. Any amount is potentially significant, in principle even as little as a single organism.

Issues arising from the conceptual contexts for nuclear and biological materials introduce another set of differences.

Inherent in the concept of "diversion" (as we know it for nuclear materials) is a time delay between the disappearance of material and a consequent malicious act that uses it. For example, diverted nuclear materials generally do not pose an immediate threat, because there are engineering steps required to incorporate the materials into a weapon. (Here we assume that the concern is about a high-yield explosion; not merely a criticality event.) It is not obvious that the resulting notion of timeliness of detection has much relevance to biological material diversion, however. Malicious use of stolen or diverted biological materials could ensue without any appreciable delay.

Another conceptual issue is the compelling rationale for "defensive research" involving

biological materials. The consequences from the malicious use of biological agents can be mitigated—the event itself takes time to develop, and measures can be employed to counter the damage. The spread of an agent through infection can be arrested by prophylactic treatments, vaccination, and quarantine. Victims can be treated. There is no analog for research on vaccinations and treatment for nuclear weapons. Were a nuclear explosion to happen, its major consequences are immediate; response to mitigate consequences is not a plausible strategy.

3.3. Implications for Material Nonproliferation Approaches

Nuclear safeguards depend on material accountancy. That is, a physical inventory of material within a designated "material balance area" (MBA) can be tracked quantitatively. Inventories can be taken by identifying and counting items, and performing analytical measurements by both destructive and non-destructive assay. The initial physical inventory for a given MBA, adjusted for increases (from material production and shipments in) and decreases (from material consumed or destroyed and shipments out), should be equal to the ending physical inventory. The problem then becomes one of explaining discrepancies, or "material unaccounted for" (MUF).

However, the differences between biological and nuclear materials have profound consequences when devising a corresponding approach for biological "safeguards." That a vanishingly small quantity of a biological material can be significant (even one organism is potentially significant since it might be grown into unlimited quantity) renders bulk accountancy meaningless. There is no practical way to deal with discrepancies. From a proliferation perspective, small, undetectable discrepancies could be critically important, while huge discrepancies paradoxically might not be significant at all.

If it were practical to always keep biological materials within sealed containers, then item accountancy might be applied. Nevertheless, even item accountancy would be a formidable task. Typically there can be millions of items, and all subject to high turnover. They would likely need to be indistinguishable items, rather than each being uniquely identified. (They might be indirectly distinguishable by context—according to a particular storage location, for example.) The integrity of each item is important, but it would be extremely difficult to apply or verify containment seals. Such is especially true when having to contend with biological materials in hosts. Although an animal might be an "item," there is no reliable containment that provides assurance of non-diversion of agent from the item.

Consider the following question for nuclear materials: What if any amount of material were significant? Imagine if one could "grow" a nuclear weapon from just a few atoms of plutonium. The implications for safeguards would be enormous. Clearly there would continue to be a need for physical protection of materials and facilities, with associated access control and screening of authorized personnel. Assays of materials could still employ qualitative ("attribute") measurements, but there would be no value to the determination or verification of material quantities or material balance. Containment and surveillance would be even more important, as would measures for the detection of undeclared activities such as the International Atomic Energy Agency (IAEA) Safeguards Additional Protocol.[4] Unfortunately, it is not clear whether these measures would be sufficient.

4. Biological MC&A

If we were to ignore the realities and argue that material accountancy should be attempted for biological materials, then any such measures are likely to be cumbersome, ineffective, and met with frustration and skepticism by the bioscience community. The danger of such an approach would be to inflict an unnecessary burden on this community, which could impede beneficial research and create an illusion of security where there is none.

An equally unsatisfactory alternative is to do nothing. Even though biological materials are not readily subject to quantitative accounting does not mean that other options cannot be found. We should not prohibit working with dangerous agents because we cannot do accountancy.

Fortunately there is a practical solution, but it requires that we modify the concept of MC&A to make it workable for biological materials. The key step is to replace "accountancy" with "accountability," as described in Section 4.3 below.

4.1. Material items

It is essential to aggregate material into items for reference and to facilitate record keeping,

while understanding that the quantity of material represented by a particular item is not necessarily meaningful. The concept of an "item" as used here must be generalized. It is not necessarily always the same as its use in (nuclear-material) item accountancy. Items are perhaps best understood as "line items," i.e., the physical entity referred to by a particular information record. Clearly a sealed vial can qualify as an item, but an access-controlled storage freezer might also qualify as an "item."

Item aggregation is accomplished both by containment and by confinement. Containment is the use of some vessel or enclosure for the biological materials. Confinement comprises the work practices that prevent materials from being taken outside of designated areas. (The only exception is the special case of a material transfer, for which special security measures are taken.)

Once materials have been aggregated into items, it is possible to gather and record information about these items. Descriptive material *attributes* characterize the biological agent or strain, its origin, date acquired, etc. The attributes can include the material type or classification, which reflects the assigned biosecurity significance. The attributes need not be unique. The material *identification* specifies a particular item of the material, e.g., the type of container, how it is identified, and where it is located. The material identification also includes a field to designate an accountable individual.

4.2. Control

For MC&A without accountancy, control becomes relatively more important. Control is one of two types. One type is an engineered or physical control, such as containment. The other, administrative control, refers to procedures that are relevant to biosecurity. Procedures need to be established for all normal conditions, such as storage, use, and common changes—creation, modification, destruction, and transfer. As much as possible, procedures should also anticipate a variety of abnormal conditions, such as inventory discrepancies, anomalies, or accidents.

4.3. Accountability

Accountability, rather than accountancy, is the critical modification to MC&A for laboratory biosecurity. Accountability assures the complete and timely knowledge of what materials exist, where they are, and who exercises control over them. All biological materials of concern must be assigned to an associated accountable person. There are no "orphaned" materials, those without an accountability assignment.

Accountability entails knowing the material status, how it is used, who is working with the material, and the context and justification for such work. It also assumes the ability to exercise control and oversight, such as protecting the material by limiting availability and access. Such access control goes beyond the level of control provided by laboratory physical security. An accountable person maintains documentation, provides reports, and responds to audits. Continuity of oversight is another essential responsibility: any accountable individual must ensure transition of that responsibility to others upon reassignment, termination, or extended absence.

In the case of accountancy-based safeguards, ultimately the materials themselves are all-important. It does not matter whether any person has been assigned particular responsibility for material items. The state has the associated accountability for all declared materials. Any further delegation of accountability is effectively irrelevant to an international safeguards approach based on material accountancy. In laboratory biosecurity, however, the materials cannot be so easily disassociated from the people who work with them.

5. Lessons for Nuclear Safeguards

We have described the essential features of a recommended implementation of MC&A for biological materials, which is modified to deal with the key differences posed by biological materials. Although one could stop at this point and conclude that we now have approaches to deal with the problem of potential proliferation of both nuclear and now biological materials, it is instructive to revisit the nuclear MC&A problem from a new perspective. Are there elements of biosecurity MC&A that could be adapted for the nuclear case? What would be the implications of an accountability-based MC&A component for nuclear safeguards?

The conventional practice of accountancy-based safeguards is a major resource drain. The implementation of complementary measures, such as the Additional Protocol, has become relatively more important. Any restructuring that can at least preserve assurances of non-diversion of declared

materials while offering resource savings is valuable, and offers the possibility of efficiencies in moving to Integrated Safeguards.

For example, could an accountability-based MC&A system be a workable and effective approach for at least some nuclear materials, such as indirect-use materials? Accountability-based MC&A is essentially just a matter of adjusting the "graded approach" to nuclear safeguards. It would retain accountancy of fissile inventories for direct-use materials, but not worry about maintaining strict inventories of (for example) natural or low-enriched reactor fuel. Such an approach would likely have consequent savings in effort to perform assays, check seals, maintain surveillance, etc. Such a system would nevertheless resemble existing safeguards—a state declares materials and facilities, and independent inspections periodically verify the declaration. Verification elements would confirm facility design information, examine operator records and reports, perform selected checks of item accountability, and conduct selective qualitative assays of material. The system would emphasize the detection of anomalies, rather than (inventory) discrepancies.

From time to time it is worth considering new approaches to familiar problems, rather than incur a risk from continuing to do things out of habit. Our safeguards practices should be subject to our own skepticism and scrutiny. In fact, this has been the case all along—safeguards have been an evolutionary development over the past sixty years. The necessity of verifying completeness of a state's declaration, and not just correctness, was a revelation more than twenty years after the Treaty on the Nonproliferation of Nuclear Weapons (NPT) and IAEA Information Circular 153.[5] So revalidating our methods, and revising when necessary, is consistent with the evolutionary development of safeguards.

Accountability-based MC&A does carry a risk, however. We would be tinkering with a system that arguably has worked over the last few decades. The effectiveness of accountability-based MC&A is untested. The details would need to be worked out, and a detailed cost/benefit analysis would be necessary to establish the rationale for accountability-based MC&A. These ideas warrant further study, consideration, and discussion by the nuclear safeguards community, especially as the MC&A approaches developed for laboratory biosecurity evolve and gain experience.

6. Summary

Biological materials pose different issues for safeguards than nuclear materials. MC&A is an important component of laboratory biosecurity, but it differs from the nuclear material precedent. MC&A modifications for biosecurity could be applicable to some aspects of nuclear safeguards. Such modifications might result in better resource utilization, but the risk of revising existing approaches must be considered.

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8. References

[1] Gaudioso, Jennifer and Reynolds M. Salerno. "Biosecurity and Research: Minimizing Adverse Impacts." Science. Vol. 304. 30 April 2004.
[2] "Biotechnology Research in an Age of Terrorism," Committee on Research Standards and Practices to Prevent the Destructive Application of Biotechnology, National Research Council of the National Academies, Washington DC, 2004.
[3] http://www.cdc.gov/od/sap/docs/salist.pdf
[4] Information Circular 540 (corrected), International Atomic Energy Agency, Vienna, 1998. See: http://www.iaea.org/Publications/Documents/Infcircs/1998/infcirc540corrected.pdf
[5] Information Circular 153 (corrected), International Atomic Energy Agency, Vienna, 1972. See: http://www.iaea.org/Publications/Documents/Infcircs/Others/inf153.shtml

Proliferation Resistance Characteristics for Civilian Nuclear Fuel Cycles Assessments

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Abstract:

This paper aims at presenting a representative sample of previous proliferation resistance assessment studies; a criterion adopted for the identification of the studies to be considered has been that of selecting only studies available in open literature. The paper will briefly review in a critical way some of the past studies carried out in this field, concentrating on the identified sets of characteristics and metrics to be used to characterise the proliferation resistance of Nuclear Energy Systems. The matter is also relevant to the PR&PP Expert Group initiative to which JRC collaborates.

Keywords: Proliferation Resistance Assessment, Decision Analysis.

1. Introduction

Since the early seventies, a lot of effort has been put in trying to evaluate the proliferation resistance of nuclear power systems and their associated nuclear fuel cycles. Past studies e.g. [1] put in evidence how it wasn't possible to conceive a proliferation-free nuclear fuel cycle, but also stressed that not all the available options are equally prone to be used as a starting point for the development of a military nuclear program. During the last thirty years several studies tried to tackle the problem of defining a suitable and comprehensive set of characteristics/attributes in order to evaluate efficiently and accurately the proliferation resistance of a given nuclear fuel cycle. This kind of analysis is useful not only in a decision making scenario related to the exportability of a given nuclear technology, but also for highlighting which are the inherent vulnerabilities of each option with respect to the proliferation threat.

Recently, interest in this field has been renewed with the need to cope securely with the disposal of excess weapon plutonium coming from dismissed nuclear weapon devices in appliance of US/Russia disarmament treaties [2,3]. In addition, the Generation IV International Forum (GIF) [4] considered the proliferation resistance of the reviewed nuclear technologies [5] as one of the key evaluation factors. Both frameworks put in evidence how the definition of such set of characteristics is of paramount importance and far from being definitively solved. The matter is also tackled by the IAEA initiative INPRO [6,7].

Rather than attempting a comprehensive overview of all the methods developed during the last decades, this paper aims at illustrating a representative sample of the different types of approaches; another criterion adopted for the identification of the studies to be considered has been that of selecting only studies available in open literature. The different approaches can be roughly classified in approaches based on a proper decision analysis framework, and in barriers approaches aimed at evaluating the barriers to the proliferation within a nuclear energy system.

Within these constraints, this paper will briefly review in a critical way some of the past studies carried out in this field [8-16], concentrating on the identified sets of characteristics and metrics to be used to

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characterise the proliferation resistance of Nuclear Energy Systems. The matter is also relevant to the PR&PP Expert Group of Generation IV International Forum (GIF) initiative to which JRC collaborates.

2. A methodology for the Assessment of the Proliferation Resistance of Nuclear Power Systems

Development context: US concerns about the dissemination of nuclear technologies to Non Nuclear Weapon States (NNWS). The time and context is that of the INFCE international and NASAP national US, studies.

Goal: To develop of a logical framework to be used by experts and/or decision makers in order to evaluate the impact of a given nuclear power system on a State's nuclear proliferation strategy.

Identified measure of effectiveness: Least proliferation resistant pathway.

Method: Multi-attribute decision analysis of selected pathways evaluated upon characteristics and sub-characteristics.

The "Methodology for the Assessment of the Proliferation Resistance of Nuclear Power Systems" study [8] was aimed at evaluating how the nuclear proliferation strategy of a proliferating State might take advantage from the availability of a particular nuclear power system. The proliferation issue is modelled as a double decision problem. The first one is a decision problem from the point of view of the international community that has to select the exportable nuclear power system from a group of possible candidates to a country. The second problem is the choice by the country, as potential proliferators, with given aspirations, of the most suitable proliferation strategy (pathway). The scheme of the methodology is illustrated in figure 1.

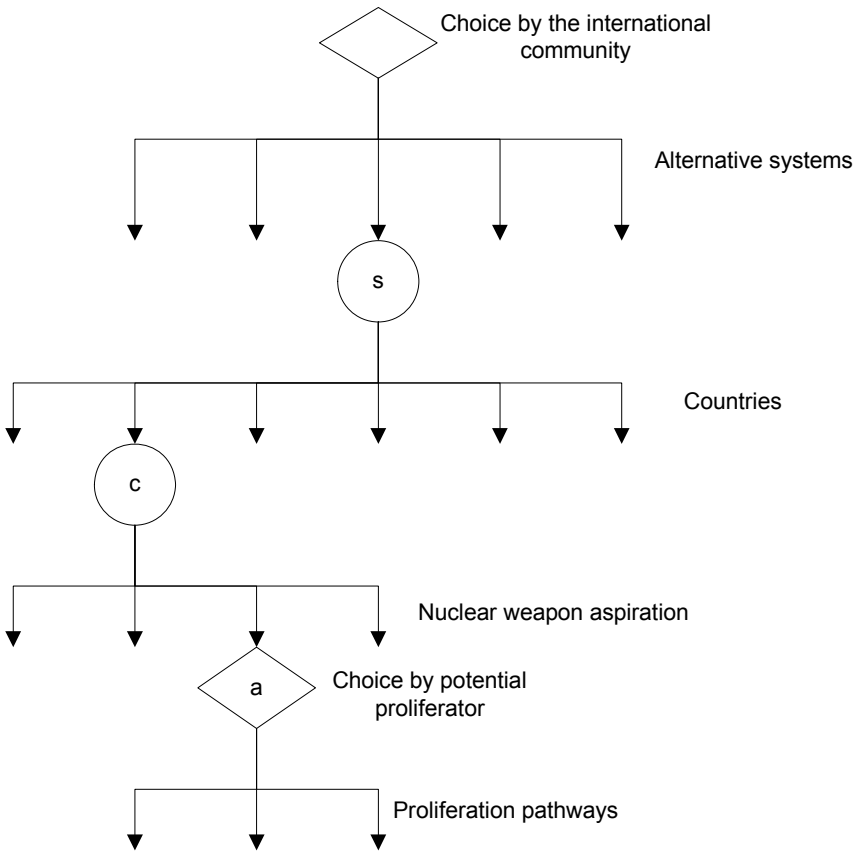

Figure 1: Scheme of the Structure of the Assessment Methodology. Adapted from [8].

The starting point of the study is the choice to be made by the international community among various candidate systems. Each of these systems (**s**) might be acquired by several different Countries, and each Country (**c**) has the possibility to choose between several possible nuclear weapon aspirations (**a**). The aspiration is defined as the quality and quantity of nuclear weapons to be made available within a given time.

Once a nuclear weapon aspiration has been defined, the Country will have to decide, among the possible proliferation pathways, which is the best one for the success of the proliferation effort.

In this way the methodology would assess how the selected nuclear power system (**s**) will affect the success of the proliferating effort of a selected hypothetical Country (**c**) with a given nuclear weapon aspiration (**a**).

Another more complex analysis might be done by considering all the possible combinations of a Country's aspiration and analysing which is the likelihood that a given Country owning the selected nuclear power system (**s**) and having a defined nuclear weapon aspiration $\mathbf{a}_j$ will be able to obtain a nuclear weapon capability $\mathbf{a}_i$ compatible with aspiration $\mathbf{a}_j$.

In order to reach the above described goals, the methodology needs to characterise the proliferation *pathways* available to a potential proliferating Country in terms of their *proliferation resistance*.

The study selected five main attributes against whom the proliferation pathways have to be characterised. After having characterised each proliferation pathway against these attributes, the least proliferation resistant pathway is chosen as an index of the proliferation resistance of the considered nuclear power system (**s**) against the selected Country (**c**) with a defined nuclear weapon aspiration (**a**).

In general the analysis will provide a set of non dominated pathways, i.e. pathways that cannot be ranked against each other unless a trade-off is accepted. In order to rank these pathways MAU theory could be used. To this aim, attributes and sub attributes are defined, together with utility functions associated to the sub-attributes.

The attributes considered by [8] for characterising the pathways are the following:

- *Weapon development time*: the amount of time required to build the first nuclear weapon device, counting from the first proliferation action. The first proliferation action is to be intended as "an action or step which is not necessary for the functioning of the alternative system as a power system regardless of whether such action is illegal or not" ([8], p. 36).
- *Monetary cost*: the total cost of the proliferating strategy.
- *Inherent difficulty*: this attribute takes into account the difficulty that a proliferating actor encounters when having to convert the available fissile material into a ready to use weapon material. It takes into account a) the scientific and technologic complexity of the process, b) the scientific personnel requirements and c) organisational and management sophistication.
- *Weapons material*: this attribute takes into account the material quality obtained at the end of processing, and takes into account the difficulty to fabricate a nuclear weapon device responding to the aspiration goal using the available material.
- *Warning Period*: it is defined as the fraction of work that still has to be done in order to achieve the selected proliferation goal when the activity is detected.

The choice of the five main attributes has been made in order to be:

1. Complete: they have to cover all the relevant aspects of the problem.
2. Operational: they have to bring useful information to the decision maker.
3. Non redundant: they have to avoid any double counting of the systems' characteristics.
4. Minimum size: they have to be the least possible number that still permits to satisfy the first requirement.

Some of the above attributes are further divided in sub-attributes, as illustrated in Table 1, which reports also the measurement units for the attributes and the associated utility functions.

Attribute	Sub-Attribute	Sub-sub-attribute	Measurement Unit & Range	Utility Functions
Weapon Development Time	Preparation (time)	-	Time (years, range 0-6)	Continuous, dependent on aspiration level, (country) and environment (e.g. crisis). 0 if 0 years -1 if ≥ 6 years Concave monotonic function
	Nuclear Material Acquisition (time)	-		
	Nuclear Material Processing (time)	-		
	Nuclear Material Fabrication (time)	-		
Monetary Cost	Direct Capital and Operating Costs	-	Money (M$, range 0-400/700)	Continuous, dependent on aspiration level, (country) and environment (e.g. crisis). 0 if 0 M$ -1 if ≥ 400/700 M$ (depending on aspiration) Convex monotonic function
	Cost incurred due to nuclear energy misuse	-		
	Nuclear fuel cycle related costs resulting from sanctions	-		
Inherent difficulty	Chemical separation of fissile material	Status of information	Dimensionless categorical scale (9 categories: A-I; depending on country science and technology level combinations	Discrete 0 if cat A: State demonstrated to master necessary (know-how) ~ -1 if cat I: State's know-how is unknown/classified Fitted by S- Shaped monotonic function
		Radioactivity	Rad/hr	Continuous 0 if 0 rad/hr -1 if 10^6 rad/hr S-Shaped monotonic function
		Criticality problems	Dimensionless (binary: high / low)	Discrete 0 if low -1 if high
	Isotopic separation of fissile material	Status of information	Dimensionless categorical scale (9 categories: A-I; depending on country science and technology level combinations	Discrete 0 if cat A: State demonstrated to master necessary (know-how) ~ -1 if cat I State's know-how is unknown/classified Fitted by S- Shaped monotonic function
		Radioactivity	Rad/hr	Continuous 0 if 1 rad/hr -1 if 10^6 rad/hr S-Shaped monotonic function
		Criticality problems	Dimensionless (binary: high / low)	Discrete 0 if low -1 if high
Weapons material	-	-	Dimensionless categorical scale (4 categories: RG Pu, WG Pu, HEU-233, HEU-235).	Discrete, depending on aspiration level. 0 for ^{235}U, -1 for RG-Pu
Warning Period			Fraction of the task that still remains to be done for proliferation to be completed, when detected. Might be expressed in terms of time fraction.	Continuous, distinguishes two regions[1]. In both regions: 0 if warning period is 100% (nothing has been done when detected -just begun) 1 if warning period is 0 (operation completed when detected).

Table 1: Attributes considered by [8].

[1] Region 1: From beginning of proliferation to the production of the first weapon; Region 2: From the beginning of the production of the first weapon to the availability of the complete arsenal.

Outcomes

A set of systems are assessed for different countries and aspiration. To this aim the attributes are estimated for a number of pathways. For each system, the least resistant pathway represents the resistance of a given system to the proliferation by a country of given aspirations.

As an example of the application of the methodology, three different systems are considered (S-I LWR, once through uranium cycle; S-II LWR, with de-naturated thorium cycle and S-III LWR with Pu recycle). A fully dedicated clandestine pathway is also taken into account. For each system, each of the identified pathways allows to fulfil the selected aspiration.

All the proliferation pathways relying on declared facilities considered by the study provide RG Pu; WG Pu is considered only within the clandestine pathway.

All systems assume the availability inside the countries of reactors only, with fresh fuel supplied from abroad and spent fuel stored in the country; light sanctions can be applied.

In the example two countries (B and C), both non weapons states, with different capabilities (B and C, both developing countries, with B more advanced than C), with two different aspirations are considered (a1, one crude, non deliverable, nuclear explosive; a2 an arsenal of ten military quality deliverable weapons to be achieved in one year from the first weapon). The possible combinations of the identified systems, countries and aspirations result in twelve different scenarios to be analysed (e.g. Country B, system S1 and aspiration a1). Each scenario is evaluated for up to ten different pathways (e.g. covert preparation-covert diversion and processing of LWR spent fuel).

The illustrative assessment is mainly carried out by comparing the pathways attribute estimates (measures). The depicted multi attribute utility approach is not really applied for the purposes of the illustrative evaluation.

3. An Approach to Quantitative Assessment of Relative Proliferation Risks from Nuclear Fuel Cycles

Development context: Study on the feasibility to quantify the nuclear material diversion risk associated to different nuclear fuel cycle options.

Goal: To quantify diversion risk.

Identified measure of effectiveness: Diversion risk of the considered nuclear fuel cycle, intended as the time integral over all the process utilities representing the diversion risk associated to each nuclear material available in the cycle.

Method: Multi-attribute decision analysis of selected pathways evaluated upon characteristics and sub-characteristics. In addition a fuzzy approach is outlined. The pathways are selected in order to cover the different source materials within a fuel cycle.

Silvennoinen, P., & Vira, J. [9] explored the possibility to calculate the *diversion risk* associated to a selected nuclear fuel cycle. The study considers the *source materials* available in a civilian nuclear fuel cycle. For each material, the related high level proliferation pathways are identified.

Each nuclear material available in the nuclear fuel cycle is associated with a utility function defining its *material diversion risk*. This material diversion risk is calculated referring to a set of ad hoc conceived *assessment criteria*. For each nuclear material considered the assessment criteria are estimated, together with some parameters representing the relative importance of each criterion for the definition of the utility function (*material diversion risk*). Once all the nuclear material diversion risks are obtained, it is possible to estimate the utility function representing the *overall diversion risk* for the considered nuclear fuel cycle; this is actually a time integral over all the material utility functions (material diversion risks) of the nuclear fuel cycle. In addition a fuzzy approach is exploited by the authors to evaluate the differences (if any) in the results when compared to the standard approach.

The *assessment criteria* used for calculating the nuclear materials' diversion risks are the following:

1 *Minimum cost*: This criterion is related to the minimum cost to be sustained for building a nuclear weapon device using the considered fissile material as the starting point. It is supposed that the fabrication of the device is to be done clandestinely, or, in alternative, the marginal cost of alterating a civil nuclear power programme to facilitate weapons construction is considered:
2 *Marginal cost*: This criterion investigates the marginal cost needed for modifying a civilian nuclear fuel cycle for fabricating nuclear weapon devices.
3 *Minimum time*: this attribute takes into account the minimum amount of time required by the proliferating actor for building a nuclear weapon device
4 *Detectability*: This criterion considers the detectability of the construction of a nuclear weapon device, considering also aspects such as the timeliness of detection.
5 *Divertability*: This criterion takes into account the ease of diverting the selected material from the civilian programme, both in terms of ease of accessing the material and of difficulty of accountancy of the targeted material flow.
6 *Quality*: This criterion takes into account the quality of separated fissile material for production of weapons.

The estimation of the *weights* to be associated with these criteria within a MAU is mainly obtained via pair-wise comparison. Two cases are considered: one exploiting the existing civilian fuel cycle, the other one based on a dedicated military programme.

Each source material is associated with one or more pathways. For each pathway the utility of the corresponding attribute (assessment criteria) is estimated taking into account the amount of the implied source material. The utility of attributes based on subjective estimation are judged upon the comparison with weapons grade plutonium produced with dedicated facilities (reference reactor and reference reprocessing facility).

In Table 2 the assessment criteria and their related measurement units, utility and weights are reported, M indicating the use of a dedicated military programme, C the use of civilian facilities.

Assessment Criteria	Measurement Unit & Range	Utility	Weights M	Weights C
Minimum Cost	US $ From < 1 M$ To >500 M$	0 if > 500 M$ 1 if < 1 M$	0.11	-
Marginal Cost	Fraction of civil programme cost From No additional cost to civil programme To Doubling of civil programme's cost	0 if Doubling of civil programme's cost 1 if No additional cost to civil programme	-	0.20
Minimum Time	Years From < 1 year To > 5 years	0 if > 5 years 1 if < 1 year	0.15	0.10
Detectability	Dimensionless	Judged upon reference case	0.17	-
Divertability	Dimensionless	Judged upon reference case	0.30	-
			-	0..35
Quality	Dimensionless	Judged upon reference case	0.17	0.19

Table 2: Assessment Criteria, measurement units, utilities and weights, adapted from [9].

Outcomes

The methodology has been used to analyse three LWR fuel cycle options, a) a once through fuel cycle b) a closed fuel cycle with uranium reprocessing and c) a closed fuel cycle with uranium-plutonium reprocessing.

To this aim seven different source materials are examined: 1) Enriched Uranium fuel, 2) Mixed Oxide Fuel (MOX), 3) Spent Fuel with short cooling time, 4) Spent Fuel in an interim storage (long cooling

time), 5) Spent fuel in a final repository still in operation (open), 6) Spent Fuel in a final (closed) repository and 7) Separated Pu. Starting from the first six source materials (separated plutonium is postulated not to be available in the Country) sixteen “detailed” proliferation pathways are considered, on the basis of different annual material flows. Once having examined the pathways and the related material diversion risks, a global diversion risk per each LWR fuel cycle option is derived. The latter is defined as a temporal average over a given time horizon. Operatively, for each option and for each year the analysts identify the source material resulting to be the maximal contributor to the risk, and assume its value as representative for the proliferation risk of the overall fuel cycle for the whole year.

The most attractive option for a would-be proliferator was found to be LWR fuel cycle option a), followed by c) and then b). The above ranking is valid in the case of low money discount rate; else the ordering found was c), a), b). Thus in both cases the most resistant (i.e. the less risky) fuel cycle is the one implying the recycling of uranium only (b). The authors stress the concept that any final choice should not be done mechanistically on the basis of utility calculations but should be checked by the decision maker.

The aggregation method used for obtaining the combined utility requires the independence of the criteria. According to the authors, this might be a too restrictive assumption, and therefore better suited aggregation methods should be investigated.

4. Risk Assessment of Alternative Proliferation Routes

Development context: Assessment of proliferation resistance of possible proliferation routes inspired by NASAP study concerns.

Goal: Quantitative evaluation of alternative options of nuclear weapons acquisition compared to the acquisition of other destructive devices.

Identified measure of effectiveness: Multi-attribute decision model used to compare the proliferation routes versus three other destructive material options.

Method: Multi-attribute decision analysis of selected routes evaluated upon characteristics and sub-characteristics.

Ahmed, S., & Husseiny, A.A. [10] evaluate the proliferation resistance of 11 alternative *proliferation routes* (e.g. isotope separation by centrifuge, research reactors, etc.) leading to the acquisition of a nuclear weapon device and compare them to three routes involving other destructive devices (chemical / biological weapons, fission product dispersal weapons, and Liquid Natural Gas –LNG-detonation). All the 14 selected routes are characterised by five main attributes, namely:

- *Resources*;
- *Difficulty*;
- *Cost/schedule*;
- *Risks*;
- *Weapons capability*.

Each of these attributes is further divided into sub-attributes defined as *acquisition factors*, thus being able to fine-tuning the characterisation of the identified routes. Attributes and acquisition factors are reported in Table 3. Each of these acquisition factors is associated to a utility or objective function. The multiplicity of these objective functions is reduced to a univariate objective function via multivariate decision theory. The final objective is to be able to provide a ranking of the proliferation routes under analysis on the basis of the degree of difficulty of their implementation. The model adopted is a multi-attribute utility function model, in which the attributes considered are the ones illustrated in Table 3.

The problem is treated as a classic decision problem, and tackled by means of decision analysis techniques. After having defined which are the acquisition factors of interest for the decision maker, an estimation of them is performed for each route.

While some acquisition factors are objectively measurable (e.g. Cost), other acquisition factors are "fuzzy" attributes, to be estimated via a subjective estimation on a 0-100 scale (e.g. Technological sophistication). The study doesn't specify whether the estimation of "fuzzy" acquisition factors is made by comparison against a reference proliferation route or not. After having estimated the acquisition factors, a utility is associated to each of them, and finally a global objective (utility) function for the pathway is derived by means of weights that should reflect the decision maker's attitude. The study reports an illustrative final ranking of the fourteen considered pathways (routes).

Category	Acquisition factors	Measurement Unit & Range	Utility (All continuous functions based on exponentials)	Weights k_i
Resources	Technological sophistication	Subjective (dimensionless, range 0-100)	1 if 0 0 if 100 Concave	0.3
	Facility requirements	Subjective (dimensionless, range 0-100)	1 if 0 0 if 100 Concave	0.26
	Instrumentation capability	Subjective (dimensionless, range 0-100)	1 if 0 0 if 100 Concave	0.11
	Personnel requirements	N. of people	1 no people required 0 for > 10.000 people Convex	0.22
Difficulty	Availability of information	Subjective (dimensionless, range 0-100)	1 if 0 0 if 100 Concave	0.15
	Accessibility to fissile material	Subjective (dimensionless, range 0-100)	1 if 0 0 if 100 Concave	0.08
Cost/ schedule	Cost	$ x 10^7 Range 0-100 (max 1B$)	1 if 0 0 if > 1B$ Convex	0.12
	Schedule (time to completion)	Years Range 0-20	1 if 0 0 if >20 Concave	0.115
Risks	Risks to personnel	Subjective (dimensionless, range 0-100)	1 if 0 0 if 100 Concave	0.015
	Risks to project detection	Subjective (dimensionless, range 0-100)	1 if 0 0 if 100 Concave	0.18
Weapons capability	Rate of weapon-grade fissile mass production	Weapons/Year	1 if >30 0 if 0 Convex	0.26
	Weapon reliability (note, actually it is an unreliability)	Subjective (dimensionless, range 0-100)	1 if 0 0 if 100 Concave	0.0618

Table 3: Attributes categories and related acquisition factors [10].

Outcomes

The analysis performed by the study reached the conclusion that, among the considered 14 routes, the easiest ones are the ones not based on nuclear proliferation, followed by dedicated proliferation routes. The least attractive routes are the ones considering the use of civilian nuclear reactors.

The study reports a coherent decision analysis exercise, all conducted under the point of view of the proliferating actor. Little emphasis is put in defining and characterising the considered actor without trying to study the eventual variation of utilities and weights by modifying the actor's characteristics.

This is however consistent with the objective of the case study, which is purely illustrative of the methodology and doesn't have any pretence of dependability of the final outcomes.

The study assigned "deterministic" values to the identified estimates of acquisition factors. Since PR analysis always takes into account some non deterministic "measures", the study highlights how this might not be the most appropriate choice, and should at least be accompanied by an accurate sensitivity analysis.

5. Proliferation Vulnerability Red Team Report

Development context: DOE's Office of Fissile Material Disposition (DOE/MD) studies on proliferation vulnerabilities of the options considered by the Fissile Materials Disposition Program [2, 3].

Goal: To address "technical considerations that can cause shortfalls in the proliferation resistance of the alternatives for disposing of US surplus weapons plutonium" [11].

Identified measure of effectiveness: Estimation of the proliferation barriers resistance via estimation of a set of attributes.

Method: Qualitative assessment of barriers effectiveness based on thorough technical analysis of Pu disposal options.

In the framework of the DOE studies on proliferation vulnerabilities of the options considered by the Fissile Materials Disposition Program (FMDP) [2, 3], SANDIA National Laboratories performed a study to assess the proliferation resistance of the alternatives for disposing US surplus weapons plutonium, focusing on the possible technical considerations that might cause shortfalls in the resistance of such alternatives [11]. In order to do so, a set of technological and institutional *barriers* have been identified, and the alternatives have been assessed with respect to these barriers.

The study identified three main categories of nuclear proliferation barriers:

a. *Accessibility* barriers, which take into account the difficulty of accessing and removing the plutonium from within a disposition process option;
b. *Observability* barriers, which take into account the difficulty of removing the plutonium without being detected, and which may also increase the likelihood of recapture;
c. *Utility* barriers, which take into account the difficulty to recover weapon-usable plutonium from the diverted material.

In addition, the study took into account a wide range of features and *measures* that have a role in the determination of the proliferation resistance, and grouped them in four *categories*:

- *Physical protection*: this category contributes to the *Accessibility* and *Observability* barriers, and covers the aspects of deterrence, detection, delay, and denial by the host against third parties' threats. These measures are implemented by the host nation, and are therefore ineffective if the proliferating actor is the host State.
- *Material control and accounting (MC&A)*: this category contributes only to the *Observability* barrier, and includes all the aspects linked to domestic safeguards plus material accountancy and complementary containment and surveillance implemented by international safeguards.
- *Environment*: this category contributes to the *Accessibility* and *Observability* barriers, and deals with the conditions under which the material is located when proliferating actions are attempted.
- *Material form*: this category influences all the three types of barriers and deals with the intrinsic properties of the material containing the target plutonium.

The contributions of each category to the types of barriers is summarised in table 4.

On the basis of information derived from the body of the study, table 5 presents the barriers, attributes, sub attributes and measurement units.

Each of these categories refers to intrinsic aspects (inherent to the technology adopted, or to the physics of the process) or institutional aspects (related to regulations) of proliferation resistance. In particular, *Physical protection* and *MC&A* are related to institutional aspects, and *Material form* is related to intrinsic aspects. The *Environment* category is related to both institutional and intrinsic aspects. The analysts have the possibility to show the results of the proliferation resistance assessment both in terms of the contribution of the institutional and intrinsic aspects or in terms of the contribution of the four above-mentioned categories.

Categories	Accessibility	Observability	Utility
Physical protection	Yes	Yes	No
MC&A	No	Yes	No
Environment	Yes	Yes	No
Material form	Yes	Yes	Yes

Table 4: contribution of the four categories to the three types of barriers. Adapted, from [11].

Barrier	Attributes Categories	Sub-Attributes	Measurement unit
Utility		Pu concentration	Weight % of Pu in form
		No. of steps	Dimensionless
		Complexity of steps	Dimensionless, linguistic
		Process efficiency	%
		Shielding / remote op	Dimensionless, Yes/No
		Published technology	% of published technology
		No. and skills of personnel	Dimensionless
		Lead time to prepare	Months
		Time for 1 SQ	Weeks
Intrinsic Accessibility and Observability	Material form	Radiation field	rem
		Pu concentration	% of Heavy Metal, Burnup
		Mass and size of fuel assembly	
	Environment		
Institutional Observability		Contribution to MUF by each step (function of throughput and measurement sensitivity)	kg
Institutional Accessibility	Evolving Security Environment	Threat	
		Safeguards and security readiness	
		Security systems analysis and design	
	Proliferation Resistance Shortfalls against SWS (Stored Weapons Standard)	Bulk processing facilities	
		Transportation (foreign and commercial)	
		NRC Facilities with different security policies (DOE Facilities, NRC Licensees, Security by others)	

Table 5: Barriers, attributes, sub attributes and measurement units.

Outcomes

The study conducted a very detailed assessment of the proliferation resistance of the disposition concepts developed by the FMDP program. These can be categorised in immobilization alternatives, borehole alternatives, reactor alternatives and long-term storage alternatives. All of these categories comprise one or more possible options (5 for immobilization, 2 for borehole, 9 for reactor and 4 for long-term storage). A detailed coverage of the analysis performed cannot be presented here, however some high level conclusions are reported.

On the basis that “All Pu is good Plutonium, some is better than other”, the highly detailed analysis performed by the study highlighted how disposal option vulnerability free doesn’t exist, and therefore concluded that:

1 A crucial point for enhancing proliferation resistance is to keep plutonium as inaccessible as possible;
2 Among all the analysed disposition alternatives, some stages are less proliferation resistant than the Stored Weapons Standard;
3 Intrinsic features of final forms may lead to some relaxation of institutional barriers (i.e. safeguards and physical protection), but cannot completely replace them;

Although Host diversion cannot be prevented, the study further concluded that undetected break out for rearmament can be considered to be unlikely.

6. Electrical Circuit Model for Quantifying the Proliferation Resistance of Nuclear Fuel Cycles

Development context: Studies for assessing the non-proliferation resistance of alternative Nuclear Fuel Cycle options.

Goal: To evaluate quantitatively the proliferation resistance of a nuclear fuel cycle option.

Identified measure of effectiveness: After having modelled the nuclear fuel cycle as an equivalent electrical circuit, the “electrical current” value is identified as the nuclear fuel cycle proliferation resistance index.

Method: Proliferation resistance assessment based on electrical analogy. Resistances are estimated by utility analysis of identified *barriers* and sub-items (attributes).

The study by Ko et al. [12] tries to model the nuclear fuel cycle as an electrical circuit, where resistances are associated to the various steps of the nuclear fuel cycle and the value of each resistance represents the resistance of that step to nuclear proliferation. The electromotive force of this circuit represents the motivation of the proliferating actor. As a result, the electrical current will be representative of the nuclear *proliferation risk* associated to that fuel cycle:

$$i = V/R_{eq}$$

The electrical current i, i.e. nuclear proliferation risk will increase with the proliferating actor's *motivation* (V) and will decrease with the effectiveness of *proliferation barriers* (taken into account in R_{eq}).

To each fuel cycle phase, a resistance is associated. Each resistance is made by the series of a number of sub-resistances representing the barriers to proliferation of the considered phase. After having identified and estimated the resistances for all the fuel cycle phases, an equivalent resistance R_{eq} for the overall fuel cycle is derived. R_{eq} is obtained by assuming that the electrical circuit representing the nuclear fuel cycle is made by the parallel of the resistances representing the single fuel cycle phases. The electrical circuit thus obtained will be analysed as a classic electrical circuit and finally a quantitative value for electric current, i.e. for the nuclear fuel cycle proliferation risk will be obtained.

According to the authors, when technical assessments of proliferation resistance of nuclear fuel cycles are performed, the proliferator’s motivation (represented by the electromotive force) can be considered constant and related to the Country’s political situation. In addition, it may be considered to be independent from technological aspects. The nuclear proliferation barriers (represented as electrical resistances) are evaluated via multi-attribute utility functions. The barriers considered by the study are based on the framework developed by Hinton, J.P., & al. [11]. The choice of the Hinton, J.P., & al. framework [11] as basis for the study’s barriers was driven by the fact that such framework enables the analysts to take into account *safeguardability* as well as proliferation resistance in an effective

way. Ko, W.I., & al. [12] presents a case study related to Korea specific situation. The identified barriers sub-items, and related measurement units and utilities are reported in table 6 (see table 5 for comparison).

Barrier	Sub-barrier	Sub items/description	Measurement Unit & Range	Utility
Utility	Processing time required to recover plutonium metal and its isotopes	Lead time to set up process	Months (1-6)	Continuous linear 0 if 1 month 1 if 6 months
		Time to process 1 SQ	Weeks (2-8)	Continuous linear 0 if 2 weeks 1 if 8 weeks
	Complexity of the mechanical and chemical process steps required for extraction of metallic plutonium	No. of process steps	No. (2-11)	Discrete Linear 0 if 2 steps 1 if 11 steps
		Shielding/remote operation requirement	Yes/no	Discrete Binary 0 if no shielding req. 1 if yes
Intrinsic accessibility/ observability	Environment	Environment between material and proliferating actors	Type of medium (various categories)	Discrete 0 if building 1 if rock mass
	Radiation barrier	Dose rate at 1m from the surface of material	Sv/hr (0-14)	Continuous Convex 0 if dose rate 0 Sv/hr ~ 1 if 14 Sv/hr
	Mass of diverted material for 1 SQ		KgHM (9-2286)	Continuous Linear 0 if mass 9 kgHM 1 if mass 2286 kgHM
Institutional observability	Limitation in the ability to measure nuclear material	Measurement uncertainty	KgPu (0-117.8)	Continuous Concave ~ 0 if 117.8 kgPu 1 if 0 kgPu
Institutional accessibility	Factors to increase opportunities for theft or diversion	Transport of SNM	Yes/no	Discrete Binary 0 if yes 1 if no

Table 6: Evaluation criteria (adapted from [12]).

As illustrated in the above table, the barriers are classified (coherently with [11]) in four main types, namely *utility*, *intrinsic accessibility/observability*, *institutional observability*, *institutional accessibility*. To each of these classes of barriers it is possible to associate more sub-barriers, each with its sub-items. The sub-items are associated to a resistance R_{ki}. These resistances values have to be estimated via expert judgement and multi-attribute utility functions. Each resistance R_k associated with a nuclear fuel cycle step k is therefore made by resistances R_{ki} associated to the sub-item i, and by estimating the R_{ki} it is possible to retrieve a value for the resistance R_k. The R_{ki} are expressed in terms of utilities and the corresponding weights are to be estimated by means of pair wise techniques.

Outcomes

The analysis performed by the study put in evidence how the identified index seems to be a promising method to capture the proliferation risk associated to a given nuclear fuel cycle.

Three different options of nuclear fuel cycle are evaluated in the context of Korea nuclear programme: Once through, DUPIC (Direct Use of spent PWR fuel In CANDU) and Thermal Recycle to use existing PWR with MOX fuel. The outcomes have been that the DUPIC option has a proliferation resistance comparable to the one considering a once-through strategy, and that the thermal recycle with MOX fuel resulted about three times less proliferation resistant than the other two alternatives.

The three fuel cycle options have been analysed with three different values of “electromotive forces” (representing different levels of motivation), and no difference in ranking occurred. In addition, even the relative proliferation resistance of the three nuclear fuel cycle options saw no relevant changes.
These observations lead to the conclusion that, apart from the absolute values, the model adopted for quantifying the proliferation resistance of nuclear fuel cycles alternatives seems to provide rankings that are not dependent on the electromotive force value.

7. Attributes of proliferation resistance for civilian nuclear power systems

Development Context: U.S. Dep. Of Energy, Nuclear Energy Research Advisory Committee (NERAC) task force on Technological Opportunities for increasing the Proliferation resistance of global civilian nuclear power Systems (TOPS) [13].

Goal: To provide a framework to assess/rate proliferation resistance of civilian nuclear fuel cycle options.

Identified measure of effectiveness: assessment of the non-proliferation barriers existing in nuclear fuel cycle facilities.

Method: Qualitative assessment of barriers effectiveness based on thorough technical analysis of nuclear fuel cycle phases.

The origin of this framework was the NAS study concerning different reactor options for disposition of excess plutonium [3]. In order to compare different solutions for Pu disposal, a technique was developed aiming at the characterization and comparison of the options via a set of *attributes*. The study identified and characterized the elements of two groups of barriers against proliferation (*intrinsic to the form of the material* and *dependent on the details of the option’s implementation*). In addition the study provided a characterization of a number of proliferating *threats* (diversion, theft, etc.).

A subsequent task force set-up within a NERAC action has further worked out the barriers concept in order to develop a suitable process for comparing the proliferation resistance among civilian nuclear power systems [14]. Three types of barriers have been proposed: *material, technical* (both intrinsic) and *institutional* (extrinsic), and a number of barriers have been identified. For each barrier a set of *attributes* for achieving a proper barrier characterization have been identified. Each attribute is then estimated and barriers rating is provided according to a qualitative ordinal scale defined for each of them on the basis of up to five classes (Ineffective, Low, Medium, High, Very High). The aim is to provide a method for proliferation resistance assessment by qualitatively estimating the effectiveness of each barrier for each fuel cycle phase, with respect to a specified threat.

In the study [14] a selected sample threat space is considered, made by a number of proliferating actors, namely: a) a *sophisticated state, overt* (a technological advanced state that overtly carries out a military nuclear programme), b) a *sophisticated state, covert* (a technological advanced state that covertly carries out a military nuclear programme), c) an *unsophisticated state, covert* (a non technological advanced state that covertly carries out a military nuclear programme) and d) a *subnational group,* e.g. a terroristic organization.

Table 7, adapted from [14], illustrates the identified barriers grouped in the three categories (material, technical and institutional), their description and their effectiveness against the threats of type b and c.

Type of Barrier	Barriers	Barriers description and identified attributes	Effectiveness for Sophisticated State, Covert	Effectiveness for Unsophisticated State, Covert
Material Barriers	Isotopic	The extent to which isotopic composition is suitable for obtaining weapon-grade material **Attributes**: critical mass; degree of isotopic enrichment; spontaneous neutron generation; heat-generation rate; radiation.	Low	High
	Chemical	Difficulty of chemical processes needed to separate wanted material **Attributes**: degree of technical difficulty needed to refine materials; existence of admixtures; no. of separate processing steps needed to obtain final material; availability of necessary processing techniques.	Very Low	High
	Radiological	Difficulty of handling and processing of material due to its radioactivity **Attributes**: specific dose rate or time required to accumulate a significant dose; need of remote handling.	Low	Moderate
	Mass and Bulk	Quantity of material that has to be processed/stolen for reaching the goal **Attributes**: concentration of material; easiness of concealing; easiness of transport.	Low	Low
	Detectability	The degree to which materials may be detected thanks to their characteristics (eg. Radiological signature) **Attributes**: degree to which material can be passively detected; degree to which active methods are necessary; hardness of radiation signatures; uniqueness of radiation signatures; uncertainties in detection equipment including screening for dummies.	Low	Moderate
Technical barriers	Facility Unattractiveness	The extent to which facilities, equipment and processes are resistant to production of potential weapon usable material **Attributes**: complexity of modifications needed for potential weapons usable material; cost of modifying facility or process to obtain potential weapons usable material; safety implications of such modifications; time required for such modifications; facility throughput; existence and effectiveness of observables associated with facility modification and misuse.	High	Moderate
	Facility Accessibility	The extent to which facilities and equipment intrinsically restrict access to fissile materials (e.g. PWR core) **Attributes**: Difficulty and time necessary to perform operations leading to access to materials, equipment and processes of concern; need for and availability of specialised equipment, skills and knowledge to gain access; extent of manual vs automatic, remote vs autonomous operation; frequency of operations potentially supporting a proliferator's end.	Low	Low
	Available Mass	The unattractiveness of a facility or an equipment is inversely proportional to the amount of nuclear material that is processed in/with it **Attributes**: amount of potentially weapons usable material in terms of critical mass.	Moderate	High
	Diversion Detectability	The extent to which diversion or theft of materials from processes and facilities can be detected **Attributes**: type of material and process involved and extent to which the process supports accurate materials accountability; uncertainties in detection equipment; form of material amenable for item counting.	Moderate	Moderate
	Skills, Expertise and knowledge	The extent to which fuel cycle-related know-how could be transferred to a military programme **Attributes**: level of specialised skills and knowledge to support specific elements of the fuel cycle; extent to which such information is directly applicable to weapons development and the applicability of dual-use skills; extent to which such information is generally available including time to fetch it.	Low	Moderate
	Time	The time that materials (and to some extent facilities and technologies) are available to potential proliferators **Attributes**: amount of immobilization time for material.	Very Low	Moderate

Type of Barrier	Barriers	Barriers description and identified attributes	Effectiveness for Sophisticated State, Covert	Effectiveness for Unsophisticated State, Covert
Institutional Barriers	Safeguards	Measures implemented to assist in the monitoring, detection, and deterrence of facility misuse and/or of material diversion or theft **Attributes**: availability of and access to relevant information; minimum detectability limits for material; existence of conspicuous signatures and ability to detect illicit activities; response time of detectors and monitors; existence, precision, and frequency of material and process inventory and control procedures; incorporation of safeguards measures into facility and process design and operation.	High	High
	Access Control and Security	The extent to which access control and physical security is a deterrent for diversion of material or misuse of facility **Attributes**: administrative steps necessary to obtain access; physical protection and security arrangements; existence of effective backup support; effectiveness of access control and security implementation.	Low	Moderate
	Location	Site remoteness may make a facility harder to attack, but it can also make it difficult to defend and increase the defenders' response time **Attributes**: not identified.	Very Low	Low

Table 7: Non-proliferation barriers description and their effectiveness with respect to selected threats. Adapted from [14].

Outcomes

The study carried out by the NERAC task force provides a detailed framework for the characterization of barriers to proliferation vs. fuel cycle phases and a possible characterization of some sources of proliferating threats.

To characterize the effectiveness of each barrier, a detailed qualitative procedure is given to be applied to any fuel cycle facility. Moreover the characterization of the threats sources provides a way to modulate the effectiveness of the barriers. For example while for a sophisticated state material barriers are of low effectiveness, they are valuable for unsophisticated ones.

The report has identified both barriers and attributes. Deliberately, there is no attempt to have coherent scales for all estimations of barriers effectiveness; moreover there is no attempt to aggregate the estimates for the different barriers within a unique proliferation resistance assessment index.

The approach was applied in the context of a comparison of nuclear energy systems [15]. In this context a number of nuclear energy systems have been considered. For each system, a number of estimates of the barriers effectiveness have been performed, i.e. different assessments are carried out for different types of proliferating actors. It is not clear if a final version of the study [15] was published.

8. An Evaluation of the Proliferation Resistant Characteristics of Light Water Reactor Fuel with the Potential for Recycle in the United States

Development context: DOE's Advanced Fuel Cycle Initiative (AFCI) to perform research leading to advanced fuels and fuel cycles for current nuclear power systems.

Goal: To evaluate the proliferation resistance associated to different options for closing Korean LWR fuel cycle.

Identified measure of effectiveness: Proliferation resistance index obtained via multi-attribute utility assessment.

Method: Assessment of *measures* used to characterise the nuclear material as it undergoes changes through the nuclear fuel cycle. Conclusions on the fuel cycle options are based on a multi-attribute utility analysis.

US Department of Energy launched an investigation aimed at identifying the possible closed fuel cycle alternatives, involving current nuclear reactors that might be used in the near future. The need of new fuel cycle designs other than the open fuel cycle currently adopted in the USA is driven mainly by two factors:

- The need of having the Yucca mountain repository to last as long as possible;
- The need to burn excess plutonium coming from dismissed nuclear weapon devices.

The AFCI Blue Ribbon Panel reviewed a number of different options for fuel to be burnt in current LWRs. Among the others, the following fuel characteristics were to be evaluated [16]:

- *Constituents* required in the fuel (fuel composition);
- Intrinsic proliferation resistance level;
- Related fabrication difficulty;
- Reprocessing and potentially increased re-fabrication difficulty;
- Suitability of fuel to be adopted by commercial nuclear power plants.

The Study [16] evaluated the proliferation resistance of a number of nuclear fuel cycles (PUREX/MOX, UREX, DUPIC, IMF –Inert Matrix Fuel-), each one involving the use of the so called "Series One" fuel[2], and compared their proliferation resistance with the Spent Fuel Standard.

The assessment relied on a multi-attribute utility assessment methodology developed at Texas A&M University by W.S. Charlton. A brief description of the methodology can be found in [16] Appendix C by W.S. Charlton.

The methodological approach chosen by the study concentrates on evaluating the proliferation resistance of the nuclear material as it undergoes changes through the nuclear fuel cycle phases. This approach leads to consider the diversion of nuclear material as the main proliferation threat associated with a civilian nuclear fuel cycle. The assessment relies on an overall proliferation resistance index (total nuclear security measure) related to the definition of five *measures*: 1) Attractiveness Level, 2) Concentration, 3) Handling Requirements, 4) Type of Accounting System, and 5) Accessibility. The measures are then broken down in a total of fourteen *attributes*. To each attribute a relative weight is associated on the basis of consultation with experts by means of written questionnaires. The measures, attributes and associated weights are reported in table 8.

Measure	Attribute x_i	Measurement Unit	Utility	Weights w_j
Attractiveness Level	DOE attractiveness Level	Dimensionless	Discrete, from DOE Attractiveness Level : 0 if category IB through 1 if category IVE	0.10
	Heating rate from Pu in material	Watts/kg	Continuous, 0 if no heating ~ 1 if same heating of ^{238}Pu	0.05
	Weight fraction of even Pu isotopes	Dimensionless	Continuous 0 no even Pu isotopes ~ 1 if only even Pu isotopes	0.06
Concentration	Concentration	SQs/MT	Continuous Approaching 0 for pure Pu, 1 if concentration < 0.01 SQs/MT	0.10

[2] Mixture of the isotopes of uranium, plutonium, neptunium, and eventually other constituents. It is supposed to be involved in a closed fuel cycle, with recycling of plutonium and possibly of other actinides. It is supposed to be used in currently available LWRs.

Measure	Attribute x_i	Measurement Unit	Utility	Weights w_j
Handling requirements	Radiation dose rates	rem/(hr SQ)	Continuous 0 if dose rate≤ 0.2 1 if dose rate ≥ 600	0.08
	Size/weight	ft^3 or lbs	Discrete 1 if size > 2 ft3 or weight > 200 lbs 0 else	0.06
Type of accounting system	Frequency of measurement	n. of measurements in a given time	Discrete From 0 if never measured To 1 if continuous measurements	0.09
	Measurement uncertainty	SQs per year	Continuous 1-x if x ≤1 0 if x > 1	0.10
	Separability	Fuel form	0 if Solid fuel with structural materials 1 if Pu/HEU metal solid	0.03
	% of processing steps that use item accounting	Fractional % of steps that use iteme accounting	0 if no item accounting 1 if 100% item accounting	0.05
Accessibility	Probability of unidentified movement	Dimensionless	~1 if 0 ~0 if 1	0.07
	Physical barriers	Dimensionless	0 if hands-on 1 if inaccessible	0.10
	Inventory	SQs	0 if > 100SQ 1 if < 1 SQ	0.05
	Fuel load type	batch or Continuous reload	0 if continuous 1 if batch	0.06

Table 8: Methodology Measures, Attributes and Weights (Charlton, in [14]).

Although the study stresses how the assessment is primarily based on material characteristics, it is possible to notice that the attributes aren't focused only on intrinsic features related to material, but take into account a broader spectrum of variables, spanning from facility operating practice to frequency of inspections.

Outcomes

The conclusion of the study was that open fuel cycles based on the so called spent fuel standard are not necessarily more proliferation resistant than closed ones, especially when considering a long time scale.

In particular the study highlights how:

- An UREX closed fuel cycle may represent a remarkable improvement in the proliferation resistance of a closed fuel cycle.
- The time integrated proliferation resistance measure of a nuclear fuel cycle aimed at transmuting minor actinides may be considered to be equivalent to the one associated to the Spent Fuel Standard.
- Recycling minor actinides has the potential of raising the intrinsic proliferation resistance of the most critical nuclear fuel cycle phases (typically enrichment/fabrication and reprocessing). This technological implementation, together with efficient nuclear Safeguards measures, has the potential to fill the gap in proliferation resistance that exists between the Spent Fuel Standard and most of the closed nuclear fuel cycle options.
- The back end of a nuclear fuel cycle seems not to be the only phase raising proliferation resistance concerns. The enrichment phase seems to exhibit comparable proliferation resistance vulnerabilities.
- A lot of research and development is been performed in order to develop advanced and innovative nuclear Safeguards measures for closed nuclear fuel cycle options. In particular, the efforts connected to the development of nuclear Safeguards for UREX fuel cycle seem to be promising.

9. General comments on the reviewed studies

Although all the reviewed studies relate to nuclear proliferation resistance issues, they don't all share a common context. The majority of the studies are aimed at characterising the proliferation resistance of different fuel cycle options[3] (Silvennoinen & Vira [9], Ko & al. [12], TOPS Attributes [13, 14], AFCI Blue Ribbon [16]). Papazoglou & al. [8] is aimed at comparing different nuclear reactors alternatives. PVRT [11] addresses the problem of the disposition options for excess weapons Pu deriving from the implementation of US and Russia disarmament treaties, and Ahmed & Husseiny [10] try to perform a comparison between a number of "standard" nuclear proliferation routes with the acquisition of other forms of non-conventional destructive weapons.

In addition, some of the studies have the objective to provide a methodology to be used -within the considered context- (Papazoglou & al. [8], TOPS Attributes [13, 14]), others have the objective to provide a dependable assessment of PR -within the considered context- (PVRT [11], AFCI Blue Ribbon [16]) and, finally, the remaining studies seem to be more an academic exercise in the PR field than any of the two former categories (Silvennoinen & Vira [9], Ahmed & Husseiny [10], Ko & al. [12]).

All the reviewed studies adopt an analysis technique somehow lent from the decision analysis paradigm, even if it is possible to identify some variations in the approaches. Indeed, the seven studies considered in this paper might fall under two broad categories of approaches: 1) Proper decision analysis approaches and 2) Barriers approaches.

Proper decision analysis approaches try to tackle the proliferation resistance problem as a proper decision problem. With the aid of a decision analysis expert, the decision maker sets (or should set…) its objective and identifies a set of attributes that he considers important for selecting the best option among a number of alternatives, all suitable to reach the identified objective. Together with domain experts, i.e. experts with knowledge of the alternatives, a set of criteria aimed to characterise properly the identified attributes are derived. After having identified a selection of alternative sequences of actions (pathways) allowing to reach the objective, domain experts estimate, on each pathway, the identified criteria, and derive an estimation of the affected attributes. By going through a decision analysis process, the decision maker will then be able to choose which might be his preferred pathway for reaching his objective. This decision analysis may rely on a full multi-attribute utility analysis.

Being the attributes chosen by the decision maker according to his needs and beliefs, there's no "perfect" set of attributes; similarly, the outcome of the decision analysis process will be entirely dependent on the decision maker attitudes, and it is not possible to state that a particular outcome is "right" or "wrong": this statement can be done only at the level of criteria and attributes estimation (which should be performed by domain experts and properly documented). For the remaining part of the analysis the only statement that can be done is whether the decision process has been *rational* or not, i.e. if it has been done coherently with the axioms of decision analysis theory [17].

This decision analysis approach seems particularly suited for taking the point of view of the proliferating actor, and the utility functions link high utility values with high proliferation attractiveness. The result is that high utility scores (where built) mean low proliferation resistance.

Among the reviewed studies, the ones adopting a proper decision analysis approach were mainly academic exercises not really aiming at producing dependable results rather to show the potentiality of the approach [e.g. 10].

Concerning *barriers approaches*, even if they can also be used in the framework of a decision analysis process [e.g. 16], they lay their roots in the safety domain, and in particular in the so called "defence in depth" concept [18]. The main concept behind a barriers approach is that, given a "threat", the system under threat can rely on a number of *barriers* that contrast this threat. The difference with safety is mainly that, in the case of safety, threats are associated to accidental initiators i.e. events happening on stochastic basis, while in the case of proliferation resistance (and physical protection) threats are associated to deliberate/intentional actions carried out be the actor (e.g. proliferators) in order to reach its objectives.

[3] The studies generally concentrate the analysis on the fuel cycles back end.

Barriers can be intrinsic to the system (material barriers, technical barriers) or provided by aspects extrinsic to the system (institutional barriers, e.g. nuclear Safeguards). Given a system and a threat (or eventually a collection of threats), the domain experts analyse the robustness of the available barriers against the selected threat(s).

The main difference between a barriers approach and a proper decision analysis approach is that, while in the latter approach the attributes are (or should be) identified by the decision maker, in the former approach barriers are (or should be) identified by domain experts.

Although the "barriers" studies share three broad categorisations of the identified barriers (utility, accessibility, observability), generally there's not a shared codified approach for drawing conclusions on the proliferation resistance of a given fuel cycle.

Studies adopting a barriers approach take the point of view of the international community, and analyse the proliferation resistance in terms of effectiveness of the identified barriers. Where built [12, 16], the defined utility functions link high utility values with high barriers effectiveness. The result is that high utility scores mean high proliferation resistance.

Among the studies reviewed in this paper, the "operative" ones (i.e. the studies that were supposed to draw dependable conclusions to be used by a stakeholder) adopted a barriers approach.

Table 9 categorizes the methodologies considered by this paper according to the above discussed aspects (context, goal, approach).

Study	Context	Goal	Approach
A methodology for the Assessment of the Proliferation Resistance of Nuclear Power Systems [8]	Comparison of different nuclear reactors alternatives.	To provide a methodology	Proper Decision Analysis
An Approach to Quantitative Assessment of Relative Proliferation Risks from Nuclear Fuel Cycles [9]	Characterisation of the proliferation resistance of different fuel cycle options	Academic exercise	Proper Decision Analysis
Risk Assessment of Alternative Proliferation Routes [10]	Comparison between a number of "standard" nuclear proliferation routes with the acquisition of other forms of non-conventional destructive weapons	Academic exercise	Proper Decision Analysis
PVRT [11]	Disposition options for excess weapons Pu	To provide dependable analysis results	Barriers
Electrical Circuit Model [12]	Characterisation of the proliferation resistance of different fuel cycle options	Academic exercise	Barriers
TOPS Attributes [13,14]	Characterisation of the proliferation resistance of different fuel cycle options	To provide a methodology	Barriers
AFCI Blue Ribbon [16]	Characterisation of the proliferation resistance of different fuel cycle options	To provide dependable analysis results	Barriers

Table 9: Peculiar characteristics of the reviewed studies.

Despite the above broad categorisation into Barriers approaches and Proper Decision Analysis (DA) approaches, the seven studies share a number of commonalities which can be summarised in the following points:

- All the methodologies identify some very high level proliferation pathways[4] on which the analysis is performed;
- While in slightly different ways, all the reviewed studies take into account both intrinsic and institutional (extrinsic) characteristics; it has to be noticed that concerning extrinsic characteristics there's no agreement among the studies on how to factor the contribution in the methodology;
- Pathways approaches share the use of characteristics related to *Technical Difficulty, Proliferation Time, Proliferation Resources* and *Material Quality*;
- Barriers approaches tend to emphasize the aspects related to *material characteristics* and *accessibility/control (observability barriers)*;
- Given a proliferator's objective the pathways are built in order to able to reach it.

Table 10 presents a mapping of some of the attributes of the reviewed PR studies based on DA versus some of the ones selected for the PR&PP evaluation Methodology [5].

Table 11 presents a mapping of the attributes of the reviewed barrier studies versus the ones selected by the PVRT Methodology [11].

Study	Technical Difficulty	Proliferation Time	Proliferation Resources	Material Quality
A methodology for the Assessment of the Proliferation Resistance of Nuclear Power Systems [8]	Yes (Inherent difficulty)	Yes (Weapons development time)	Yes (Monetary cost)	Yes (Weapons material)
An Approach to Quantitative Assessment of Relative Proliferation Risks from Nuclear Fuel Cycles [9]	Partially (Ease of material diversion)	Yes (Minimum time)	Yes (minimum and marginal costs)	Yes (Quality of separated material)
Risk Assessment of Alternative Proliferation Routes [10]	Yes (Difficulty of Route)	Yes (Cost & Schedule)	Yes (Cost & Schedule)	Indirectly (Weapon Reliability)

Table 10: DA Approaches attributes mapped vs PR&PP ones [5].

Study	Utility	Observability	Accessibility
PVRT [11]	Yes	Yes	Yes
TOPS Attributes [13,14]	Yes (Material and Technical barriers)	Yes (Institutional barriers + detectability and diversion detectability	Yes (Technical barriers/facilty accessibility)
Electrical Circuit Model [12]	Yes	Yes	Yes
AFCI Blue Ribbon [16]	Yes (Material attractiveness, concentration & handling requirements)	Yes (Type of accounting system)	Yes

Table 11: Conceptual Mapping of Barriers Approaches vs PVRT [11].

[4] These proliferation pathways shouldn't be confused with the proliferation pathways derived by studies aimed at performing Safeguards Assessments (for a review of a collection of methods aimed at this see e.g. [19]). The latter ones are built using systems analysis techniques such as Fault Trees etc.

10. Conclusions

The last fifteen years saw a renewed interest in the development of effective Proliferation Resistance assessment methodologies. The driving forces of the new effort in this field have been a) the need to identify the most effective way for disposing the excess weapon plutonium coming from the dismissed US and Russian nuclear weapon devices and b) the development of a new generation of nuclear power systems, which will have to be a substantial improvement in a number of fields, including Proliferation Resistance. In this latter framework two important international initiatives have started: INPRO [6, 7] within IAEA and PR&PP [5] within GIF [4].

Although various studies already tried to tackle the PR assessment issue, no final agreement has been reached on which are the most suited attributes for capturing the aspects affecting proliferation resistance of a nuclear fuel cycle or of part of it.

Indeed, for methodologies relying on a decision analysis approach, the identification of the attributes of interest should capture the decision maker attitude.

This paper presented a survey of Literature on Proliferation Resistance methods, concentrating on how the methodologies capture the relevant aspects for characterising the proliferation resistance of the systems under analysis (characteristics/attributes).

This survey has been performed in the context of JRC activities on the applicability of systems analysis techniques to nuclear Safeguards [19-21] and to Non-proliferation. In addition, the work will be part of JRC contribution to the PR&PP Expert Group GIF initiatives and will provide input for discussions aimed at issuing revision 3 of the Gen IV PR&PP evaluation methodology report.

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References

[1] International Nuclear Fuel Cycle Examination (INFCE), (1980). *Report of the First Plenary Conference of the International Nuclear Fuel Cycle Examination (INFCE)*, International Atomic Energy Agency, Vienna.

[2] NAS, (1994). Committee on International Security and Arms Control, National Academy of Sciences, *Management and Disposition of Excess Weapons Plutonium*, National Academy Press, Washington D.C.

[3] U.S. National Academy of Sciences, (1995). *Management and Disposition of Excess Plutonium: Reactor-Related Options*, Committee on International Security and Arms Control, Panel on Reactor-Related Options, National Academy Press, Washington, DC.

[4] US DOE, (2002). Nuclear Energy Research Advisory Committee and the Generation IV International Forum, *A Technology Roadmap for Generation IV Nuclear Energy Systems,* GIF002-00, December.

[5] PR&PP Expert Group, (2005). Evaluation Methodology for Proliferation Resistance and Physical Protection of Generation IV Nuclear Energy Systems, *27th ESARDA Annual Meeting Symposium on "Safeguards and Nuclear Material Management",* London.

[6] INPRO, *Guidance for the evaluation of innovative nuclear reactors and fuel cycles, Report of Phase 1A of the, International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO),* IAEA-TECDOC-1362.

[7] Haas, E., (2005). INPRO Proliferation Resistance Assessment Methodology, *27th ESARDA Annual Meeting Symposium on "Safeguards and Nuclear Material Management",* London.

[8] Papazoglou, I.A., Gyftopoulos, E.P., Miller, M.M., Rasmussen, N.C., & Raiffa H., (1978). A Methodology for the Assessment of the Proliferation Resistance of Nuclear Power Systems, MIT-EL 78-021, Department of Nuclear Engineering and Energy Laboratory, Massachusetts Institute of Technology, September 1978.

[9] Silvennoinen, P., & Vira, J., (1981). An Approach to Quantitative Assessment of Relative Proliferation Risks from Nuclear Fuel Cycles, *Journal of Operational Research Society*, Vol. 32 (1981) pp. 457-466.

[10] Ahmed, S., & Husseiny, A.A., (1982). Risk Assessment of Alternative Proliferation Routes, *Nuclear Technology*, Vol. 56 (1982) pp. 507-515.

[11] Hinton, J.P., Barnard, R.W., Bennett, D.E., Crocker, R.W., Davis, M.J., Groh, H.J., Hakkila, E.A., Harms, G.A., Hawkins, W.L., Hill, E.E., Kruse, L.W., Milloy, J.A., Swansiger, W.A., & Ystesund, K.J. (1996). Proliferation Vulnerability Red Team Report, SAND97-8203•UC-700, SANDIA National Laboratories, October 1996.

[12] Ko, W.I., Kim, H.D., Yang, M.S., & Park, H.S., (2000). Electrical Circuit Model for Quantifying the Proliferation Resistance of Nuclear Fuel Cycles, *Annals of Nuclear Energy*, Vol. 27 (2000) pp. 1399-1425.

[13] NERAC, (2000). *Report of the International Workshop on Technology Opportunities for Increasing the Proliferation Resistance of Global Civilian Nuclear Power Systems (TOPS)*, LLNL, March 29-30, 2000.

[14] NERAC, (2000). *Annex: Attributes of Proliferation Resistance for Civilian Nuclear Power Systems*, October.

[15] Taylor, J., & Schock, R., (2000). *Proliferation Resistance Technology Assessment of Nuclear Reactor Systems*, July 28, 2000, Draft2.

[16] Baron, P., Brown, C., Kaiser, B., Matthews, B., Mukaiyama, T., Omberg, R., Peddicord, L., Salvatores, M., & Waltar, A., (2004). *An Evaluation of the Proliferation Resistant Characteristics of Light Water Reactor Fuel with the Potential for Recycle in the United States*, Waltar, A., & Omberg, R., eds, final report, November. Available at: www.radiochemistry.org/documents/docs/1104_BlueRibbonComm.doc

[17] Keeney, R. L., & Raiffa, H., (1976). *Decisions with Multiple Objectives: Preferences and Value Tradeoffs*, John Wiley & Sons, New York.

[18] IAEA (1996). *Defence in Depth in Nuclear Safety*, INSAG-10, Vienna.

[19] Cojazzi, G., & Renda, G. (2003). Applying systems analysis and risk based approaches to assess nuclear Safeguards and non proliferation regimes, *25th ESARDA Annual Meeting Symposium on "Safeguards and Nuclear Material Management"*, Stockholm, 13 to 15 May, 2003, European Commission.

[20] Cojazzi, G., & Renda, G. (2004). *Safeguards Assessment Methodologies: Review and Perspectives from a Systems Analysis Point of View*, European Commission, EUR 21307 EN

[21] Cojazzi, G.G.M., Renda, G., & Contini, S., (2004). Qualitative and Quantitative Analysis of Safeguards Logic Trees, *PSAM-7, ESREL-'04,* Berlin.

Evaluation Methodology for Proliferation Resistance and Physical Protection of Generation IV Nuclear Energy Systems

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Abstract

This paper provides an overview of the methodology approach developed by the Generation IV International Forum Expert Group on Proliferation Resistance & Physical Protection for evaluation Proliferation Resistance and Physical Protection robustness of generation IV nuclear energy systems options.

The paper highlights the current achievements in the development of the Proliferation Resistance and Physical Protection Evaluation Methodology. The way forward is also briefly presented together with some conclusions.

Keywords: Proliferation Resistance, Physical Protection Robustness, Evaluation/Assessment.

1. Introduction.

The Generation IV International Forum (GIF) was initiated in 2000 and formally chartered in mid 2001. It is an international collective effort 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, on July 30, 2003, the GIF agreement, becoming thus the eleventh participating "Country".

Generation IV nuclear energy systems will have to exhibit highly innovative solutions and technologies requiring extensive research and development effort, hence the new concepts will be ready for deployment only in the years 2030 (Figure 1) [1].

The Technology Goals for Generation IV nuclear energy systems, developed during the Roadmap project [2], highlight Proliferation Resistance and Physical Protection (PR&PP) as one of the four goal areas in which these technologies will have to excel, along with Sustainability, Safety & Reliability, and Economics.

On the basis of these four goal areas an evaluation methodology was developed during the roadmap project [2] which contributed to identify the nuclear energy systems options to be considered for further development. At the and of the roadmap phase, six candidate Generation IV reactor concepts were selected by GIF for development together with the associated fuel cycles options. The selected reactor concepts are, in alphabetic order, Gas-cooled Fast Reactor (GFR), Lead-cooled Fast Reactor (LFR), Molten Salt Reactor (MSR), Super Critical-Water-cooled Reactor (SCWR), Sodium-cooled Fast Reactor (SFR) and Very High Temperature Reactor (VHTR).

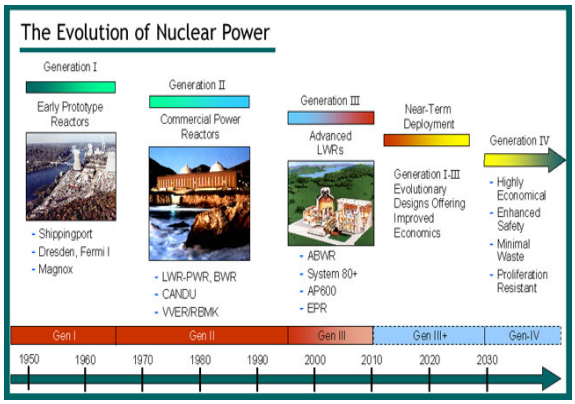

Figure 1. The evolution of nuclear power: the 4 Generations of Nuclear Reactors. [1].

The proliferation resistance of the candidate options were roughly evaluated on the basis of: spent fuel characteristics and presence or absence of separated plutonium in the associated fuel cycle.

The physical protection robustness was evaluated on the basis of the degree of passive safety features.

The Generation IV Roadmap [2] recommended however the development of a comprehensive evaluation methodology to assess PR&PP of Generation IV nuclear energy systems. Accordingly, an Expert Group was formed and tasked by the GIF in December 2002 to develop an improved evaluation methodology on the basis of the recommendation in [2]. The expert group includes members of the GIF Countries and representatives from the IAEA.

The methodology, builds innovatively on previous efforts and methods (see [3 and 4] for a review of PR methods). Some of the features of the evaluation method being developed have been already reported in previous meetings and in international conferences [e.g. 5-8].

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. After a number of iteration and internal reviews within the expert group members, the revision 2 Draft Evaluation Methodology for Proliferation Resistance and Physical Protection of Generation IV Nuclear Energy Systems methodology report has undergone to a process of internal peer review resulting, on September 2004, in the internal release of the revision 2 methodology report, representing a major milestone of the project [9]. On January 2004, an example case study, named the Development Study has been initiated with reference to a hypothetical L2 type [1], medium size Sodium Fast Reactor with on site re-fabrication of fuel elements by means of pyro-processing techniques. The aim of the development study is basically to advance with the methodology development rather to arrive at an evaluation of the L2 nuclear energy system, named ESFR (Example Sodium Fast Reactor).

In addition the evaluation methodology has been presented and debated together with Generation IV systems designers in a workshop held in Crystal City in Arlington Virginia on November 2004.

This paper will provide a full overview of the methodology approach developed by the PR&PP Expert Group according to revision 2 Evaluation Methodology for Proliferation Resistance and Physical Protection of Generation IV Nuclear Energy Systems internal project report of the PR&PP Expert Group [9].

The paper highlights the achievements already reached in the Proliferation Resistance and Physical Protection evaluation methodology, and the direction taken to tackle the issues still open.

2. Framework for the Evaluation

PR&PP evaluation can be applied across the entire fuel cycle of each Generation IV nuclear energy system, as illustrated in Fig. 2. The approach facilitates the use of information resulting from the application of the methodology to aid in decisions among alternative policy choices regarding the development of Generation IV systems. The approach also permits comparisons of designs or comparison of options that can provide information to designers on how a system can be improved. It could also permit comparison with a reference system, if one is defined.

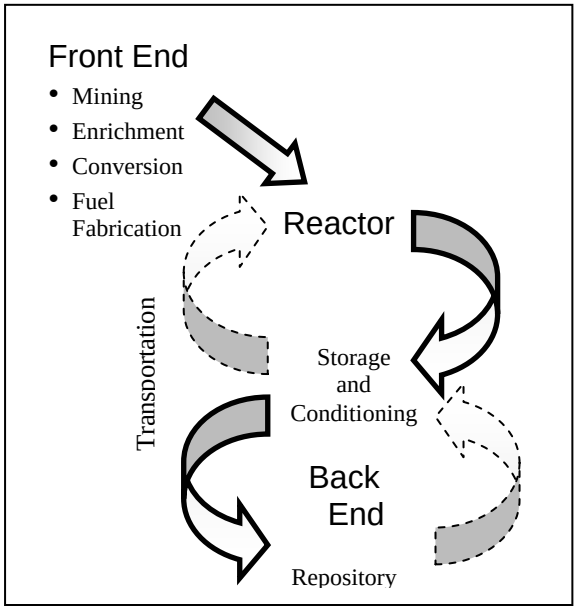

Figure 2. Main Elements of a Nuclear System.

The methodology is organized as a progressive approach under a single integrated structure, applying alternative methods at different levels of thoroughness, as more detailed design information is developed, and as research improves the depth of technical knowledge. The overall methodological approach is outlined in Figure 3.

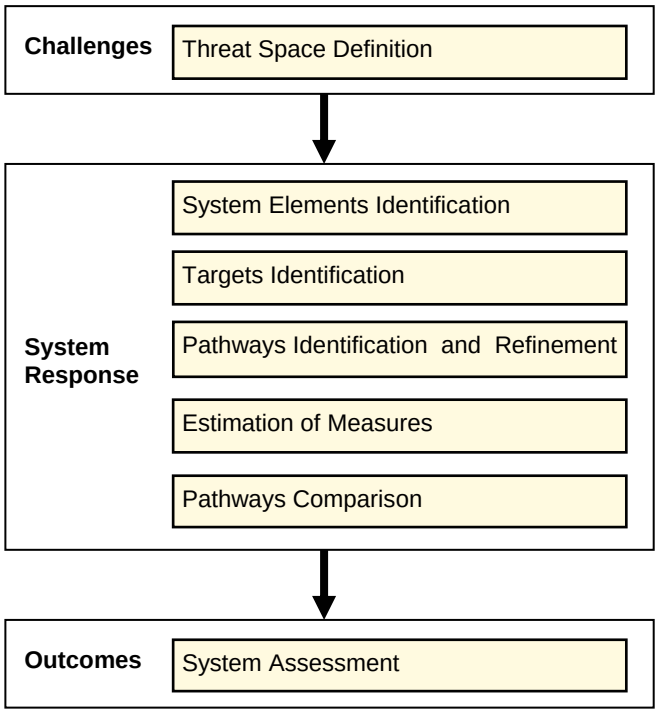

Figure 3. Framework for PR&PP Evaluation.

It considers a set of alternative systems and evaluates their resistance or robustness to a collection of potential threats. For a given system, a set of *challenges* is identified, the *response of the system* to these challenges is assessed and expressed in terms of *outcomes*. The challenges to the system are given by the threats posed by potential proliferators and sub-national adversaries on the nuclear systems.
The characteristics of the Generation IV systems, both technical and institutional, are used to evaluate their response to the threats and determine their resistance against the proliferation threats and robustness against sabotage and terrorism threats. System response encompasses five elements:

1. *System Elements Identification*. The nuclear energy system is decomposed into smaller elements (subsystems) at a level amenable to further analysis.
2. *Targets Identification*. A systematic process is used to identify targets, within each system element, that actors (proliferators or adversaries) might choose to attack or use.
3. *Pathways Identification and Refinement*. Individual pathway segments are developed through a systematic process, analyzed at a high level, and screened where possible. Segments are connected into full pathways and analyzed in detail.
4. *Estimation of Measures*. Measures are first estimated for segments and then the estimates are aggregated for complete pathways.
5. *Pathways Comparison*. Results are displayed as appropriate to permit Pathways Comparisons.

The *Outcomes* of the system response are expressed in terms of proliferation resistance and physical protection measures in order to allow a *System Assessment*.

3. Approach to the Evaluation

A progressive approach permits broad application of the PR&PP evaluation to Generation IV systems. PR&PP analysis can be applied to systems under development through fully designed systems. The scope and complexity of the assessment of the response can be appropriate to the level of detailed design information available, and to the level of detail with which the threat space can be specified. Very little specific design information exists for systems in the early development stages, and thus in-depth scrutiny of the system flows, physical plant layouts, and interdependencies cannot be probed through systematic evaluation. Rather, the methodology itself will help to guide the design evolution through a successive progression of iterations between design development and design evaluation.

The progressive system evaluation begins with the collection and organization of information and the development of coarse pathway models. Information is collected on each energy system: properties of the fuel cycle, designs of each system element, physical characteristics and institutional framework.

A representative set of threats, including characterization of the actors and strategies is defined. A formal identification process for the targets and pathways is then applied. Pathways for each target are identified and screened to eliminate those that are implausible or are clearly subsumed into other, more representative pathways. The evaluation of system response is then performed for the reduced pathway set.

In the most thorough application of the approach, the design will be far enough along to identify component characteristics, points of possible human interaction, safeguards protocols, procedures and training, and physical mechanisms that apply. Mechanistic calculations can run all the way from simple mass and energy balances to the systematic

consideration of uncertainty. It is necessary to apply judgment to such information and adapt what is available to what is needed; this transformation always results in uncertainty that needs to be considered in the analysis. Several forms exist to structure the pathway analysis (directed graphs, event tree/fault tree models, simulation models, etc.) and selection will depend on the scenarios themselves, the state of design information, the quality and applicability of available information, and the analyst's preferences.

3.1 Challenges

To evaluate either proliferation resistance or physical protection robustness, one must first specify "resistance" or "robustness" against whom (e.g., the actor), and against what actions (e.g., the strategy).

The concept of threats and threat definition has been well developed in the field of physical protection, where the characteristics of potential adversaries and their potential strategies have long been defined as a prelude to subsequent studies of physical protection system response.

For PR&PP evaluations, a detailed framework is provided for defining the set of threats, named *Threat Space*, that could potentially challenge nuclear energy systems. For both PR and PP, the threat definition includes characteristics of both the actor, and the actor's strategy.

For PR, the actor is the host State for the nuclear energy system, and the threat definition includes the proliferation *Objectives* (e.g. number and characteristics of nuclear explosives sought) as well as the *Capabilities* (skills, resources) of the State. For PR the potential threat *Strategies* include diversion, undeclared production, abrogation using declared facilities and materials, and the construction of completely separate, clandestine facilities.

For PP threats, the actor, or adversary, is an individual or group composed of some combination of outsiders and insiders. The PP adversary's characteristics are defined by the adversary's

- *Objective*, which may be either the theft of nuclear material, radioactive material or information, radiological sabotage, or sabotage to disrupt operations;
- *Capabilities*, which include their knowledge, skills, weapons and tools, number, and level of dedication, and
- *Strategy* that is defined by the adversary's mode of attack, which may range from ground-based to standoff to cyber; and the adversary's tactics, including stealth, deceit, and overt force.

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, and what threats to exclude, 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. In the evaluation of the system response to a given threat, the uncertainty in the system response is then evaluated independently of the probability that the system would ever actually be challenged by the threat, that is, PR&PP evaluations are contingent upon the challenge occurring.

The detail with which threats can and should be defined depends upon the level of detail of information available about the system design and the locations where the system would be deployed. In the earliest stages of conceptual design, where detailed information about the location of deployment is likely limited, relatively stylized but reasonable threats must be selected. Conversely, when design has progressed to the point of actual deployment, detailed and specific characterization of both the system location and potential threats become possible.

3.2 System Response

3.2.1. System Elements Identification

To analyze the nuclear energy system response to the specific threats identified, the boundaries of the system, which will limit the scope of the evaluation, must be clearly defined. Subsequently, the analyst identifies elements of the system in ways that facilitate the analysis. The term System Element is formally defined as a subsystem of the nuclear energy system; it can comprise a facility (in the systems engineering sense), part of a facility, a collection of facilities, or transportation system within the identified nuclear energy system where diversion/acquisition and/or processing could take place (PR) or theft/sabotage could take place (PP).

3.2.2. Targets Identification

The term target is defined as nuclear material, equipment, processes and information in a nuclear energy system to be protected from the PR or PP threat. Targets are the interface between the proliferator or adversary and the nuclear system; they form the link between the objectives and the system elements.

Physical protection targets are nuclear material, equipment or information to be protected from PP threats of theft and sabotage. Proliferation resistance targets are nuclear material and processes to be protected from PR threats of diversion and undeclared production.

Target identification is conducted by systematically examining the nuclear system for the role that materials, equipment and processes in each system element could have in each of the strategies identified in the threat definition. Typically, this requires iterative identification, review and revision to take different aspects of the strategy into consideration.

3.2.3. Pathways Identification and Refinement

A system pathway analysis for a set of threats consists of identification of potential sequences of events and actions that lead to the undesirable outcome (proliferation, sabotage or theft) and the evaluation of the system response. Given that there are uncertainties associated with predicting the response of the system with respect to a given set of threats, a probabilistic pathway analysis is a natural structure for assessing system response to different types of threats.

Pathways suggest a natural data structure for the analysis of a system's performance. They specify individual cases for which the measures can be evaluated. They can be regarded as the response of the system to the posed threat. Following the systems analysis format shown in Figure3, given a threat, the nuclear energy system responds through the pathways with a number of stumbling blocks, barriers and challenges, and the outcome of this will be, for each pathway, a set of specific values for the measures for that particular pathway. The pathways can be graphically represented e.g. in the form of functional event trees. The functional event tree shown in Figure 4. summarizes the main branch points of the paths to proliferation (acquisition, processing, and fabrication) and displays for each pathway the six measures of interest.

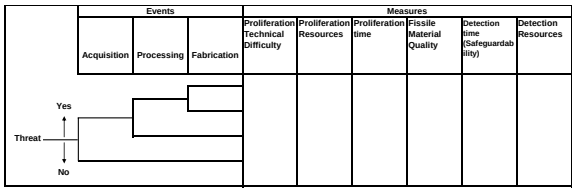

Figure 4. Functional Event Tree for Proliferation Resistance.

A similar functional event tree can be used to describe the paths to theft or sabotage.

3.2.4. Estimation of Measures

High-level measures are defined for both PR and PP. These measures are the high-level characteristics of a pathway that include information important to the evaluation methodology users and to the decisions of a proliferator or adversary. For PR the following preliminary measures were identified:

- Proliferation Technical Difficulty – The inherent difficulty, arising from the need for technical sophistication and materials handling capabilities, required to overcome the multiple barriers to proliferation. This measure does not include the technical difficulty of concealing the diversion or undeclared production; these will be reflected in proliferation resources, proliferation time, and detection time (safeguardability).
- Proliferation Resources – The economic and manpower investment required to overcome the multiple technical barriers to proliferation including the use of existing or new facilities.
- Proliferation Time – The minimum time required to overcome the multiple barriers to proliferation; i.e. the total time planned by the State for the project.
- Fissile Material Quality – The degree to which the characteristics of the material affects its utility for use in nuclear explosives.
- Detection time (Safeguardability)[1] – The time following the initiation of diversion or undeclared production, for detection

[1] Note: As a result of working on the Development Study, the PR&PP group is planning to replace this measure with *safeguardability* to capture not only the detection time, but also elements of feasibility and degree of difficulty for implementation of effective safeguards for the system. A possible definition is: Safeguardability – The extent to which a system can be effectively [and efficiently] put under international safeguards.

resources to detect irregularities and to provide adequate confirmation that diversion or undeclared production has occurred or is occurring.
- Detection Resources – manpower, technology, and funding required to apply international safeguards.

For PP, the following preliminary measures were identified:

- Operational Accessibility – the frequency and duration of access to vital equipment, systems, and zones, required for operations, surveillance, and maintenance activities performed by privileged personnel.
- Adversary Delay – the time required to overcome intrinsic barriers to accessing and disabling a vital equipment target set (sabotage) or to removing materials (theft).
- Interruption Delay – the additional delay time created by protective force response prior to neutralization.
- Detection Time – the time required, after intrusion by outsiders or unauthorized action by insiders, for physical protection system alarms to be received and verified.
- Consequences – the consequences of a failure to neutralize a threat; mitigation following the attack can control the extent of the consequences.
- Physical Protection Resources – the manpower, capabilities, and costs required to provide physical protection (background screening, detection, interruption, and neutralization) and the sensitivity of these resources to the changes in the threat sophistication and capability.

By considering the 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 surveillance. This can reduce the requirement for access to such vital equipment by plant personnel, reducing the potential for insider action ("Operational Accessibility") as well as permitting simple, passive measures to delay access to vital equipment ("Adversary Delay") to prevent sabotage.

The question of independence of the high level measures is often raised among practitioners of methodologies of proliferation resistance. There is a concern that a decision maker should be provided with information that is not ambiguous and might be presented in terms of redundant factors. Interdependencies of measures should be clearly identified in a PR&PP evaluation and addressed in the estimation of measures.

3.2.5. Pathways Comparison

A pathway analysis is performed by considering multiple pathway segments. There will be measures estimated for the individual segments that will need to be aggregated to estimate the measure for the pathways.

Although aggregation of measures for different pathways may be performed, it is in general more valuable to be able to compare the measures for different pathways and determine the relative importance of different pathways.

The objective of the system assessment is then the identification of the dominant pathways and the measures associated with them.

3.3. Outcomes and Presentation of Results

The presentation of results is an important aspect of the evaluation, especially because the results are intended for two types of users: system designers and program policy makers. Thus, the analysis of the response of the system must be amenable to being expressed in different levels of detail. Program policy makers are likely to be interested in the high level measures, while system designers will be interested in measures and metrics that are more directly related to the system design characteristics.

Over time, results will also be calculated in different forms through the progressive approach. During early phases, the methodology will provide qualitative information and identify areas of uncertainty (technical knowledge gaps), while more quantitative results will be expected later.

The results of the PR&PP assessments will be given in terms of the high level measures presented above. While the resources employed to protect against proliferation, theft, or sabotage will ultimately be subsumed in the overall cost of the nuclear energy system, it is necessary to include assumptions about resources in order to assess the time required for detection by safeguards, or the effectiveness of physical protection. An example of a "final" top level result for a program policy maker or a system designer would take the form shown in Table I for the case of proliferation resistance. A similar table can be generated for physical protection.

	Prolif. Technic. Difficulty	Prolif. Resources	Prolif. Time	Fissile Mat. Quality	Det. Time (Safeguard ability)	Det. Resources
Opt. #1						
Opt. # 2						
Opt. # 3						
Opt. # 4						

Table I. Example of a Top Level Decision Table for Proliferation Resistance.

Here four options are arbitrarily indicated to illustrate that range of choices that a program policy maker or a system designer might face. These options will be specified according to the particular question at hand. For example, for a program policy maker, the options may refer to a given country, with given capabilities and objectives. Options 1-4 could then be three different Generation IV nuclear energy systems plus a dedicated path to proliferation. Each row would represent the path offering the least resistance for the given system.

The final steps in PR&PP evaluations are to integrate the sub-elements of the analysis and to interpret the results. This includes development of best estimates for numerical and linguistic descriptors that characterize the results, development of distributions reflecting the uncertainty associated with those estimates, and development of appropriate displays to communicate uncertainties.

3.4 Implementation

The performance of a PR&PP evaluation follows a sequence of well defined steps that starts with the initial process of scope definition and ends with the final peer review of the evaluation report and its conclusions.

4. The way forward and conclusions

The development of the Proliferation resistance and Physical Protection Evaluation Methodology is a collective effort carried out by the PR&PP Expert Group within GIF. The Methodology aims at progressive implementation as more detailed design in formation becomes available and should be suitable for implementation since early design stages.

The approach is based on the paradigm Challenges, Systems Response and Outcomes and will be developed coherently for both the Proliferation Resistance and Physical Protection robustness evaluation.

It is developed within an international working group by means of a consensus approach.

This paper presented the status of the evaluation methodology as described in the revision 2 report issued in September 2004 [9]. On January 2004, an example case study, named the Development Study has been initiated with reference to a L2, medium size Sodium Fast reactor with on site re-fabrication of fuel elements by means of pyro-processing techniques. As a result of the work done during the development study, and of a workshop held with systems designers, a number of areas for further work and refinement emerged, for example:

- The adopted measures for the characterization of the PR and PP pathways. The currently adopted measures are being revised in order to assess the coverage of all the relevant characteristics of the pathways. The measures should be complete, non redundant, informative and the least possible. Moreover for each measure a suitable set of metrics (i.e. measurements units) and measurements scales should be identified, together with a suitable scheme of aggregation within the pathways segment estimates.
- The Issue of completeness and exhaustiveness of pathway identification and analysis. While the systematic identification and analysis of all possible existing pathways in a mechanistic approach is not a methodology goal an effort has to be done to identify a manageable number of representative pathways. A possible direction to be investigated is by means of a suitable categorization of the PR and PP targets.

The methodology is currently being reviewed by the PR&PP group and an update will be issued in September, 2005.

The development case will be continued and further extended to constitute a Demonstration Case. An implementation guide based on the methodology and on the demonstration case will be also developed.

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Note: the list of PR&PP expert group members is periodically updated and revised; the list here reported is consistent with that included in Revision 2 methodology report. [9].

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References

[1] http//gen-iv.ne.doe.gov/

[2] US DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum, *A Technology Roadmap for Generation IV Nuclear Energy Systems,* GIF002-00, December 2002.

[3] *Guidelines for the Performance of Nonproliferation Assessments,* Pacific Northwest National Laboratory, PNNL-14294, May 2003.

[4] S.V. Mladineo, R. Denning, R. Bari, J. Eagle, C. Olinger, G. Rochau, J. Roglans, R. Schock, Guidelines for the performance of Nonproliferation Assessments, *ESARDA 25Th Annual Meeting*, Stockholm May 13-15, 2003.

[5] B. Bari, P. Peterson, J. Roglans, and S. Mladineo, Proliferation Resistance Modeling, *ESARDA/INMM Workshop, Safeguards Perspectives for a Future Nuclear Environment*, 14-16 October 2003, Como, Italy.

[6] Peterson PF, Roglans J, Bari RA. Assessment methodology development for proliferation resistance and physical protection of Generation IV systems. *Proceedings of ANS International Conference – Global 2003*, New Orleans, LA, November 13-18, 2003.

[7] R. Nishimura, P. Peterson, J. Roglans, R. Bari, D. Kalenchuk, Development Of A Methodology To Assess Proliferation Resistance And Physical Protection For Generation IV Systems, *Americas Nuclear Energy Symposium, ANES 2004*, Miami Beach, Florida, October 3-6, 2004.

[8] B. Bari, PR&PP, A Methodology for the Evaluation of Proliferation Resistance & Physical Protection of Generation IV Nuclear Energy Systems, *2004 ANS Meeting*, November 14-18, Washington, DC, USA.

[9] PR&PP Expert Group, *Evaluation Methodology for Proliferation Resistance & Physical Protection of Generation IV Nuclear Energy Systems*, September 2004, Revision 2, Internal Project Report.

INPRO Proliferation Resistance Assessment Methodology

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Abstract:

Following a resolution of the General Conference of the IAEA in the year 2000 an International Project on Innovative Nuclear Reactors and Fuel Cycles, referred to as INPRO, was initiated.

INPRO aims inter alia at the development of a methodology for assessing innovative nuclear systems (INS) in particular with regard to economics, environmental impact, waste management, safety of nuclear installations and proliferation resistance, and to use this methodology with an internationally acknowledged IAEA set of recommendations for making such assessments.

To counter the potential misuse of an INS for the purpose of producing nuclear weapons, proliferation resistance has become an important issue for the further development of nuclear energy. For the assessment of the proliferation resistance of an INS, INPRO has defined basic principles (BP), user requirements (UR), and a hierarchical set of indicators and acceptance limits. In addition, a preliminary list of variables for the evaluation of Indicators has been determined.

INPRO has established a sound basis for the development of an internationally acknowledged proliferation assessment methodology. It may help designers to assess the impact of intrinsic design features on the overall proliferation resistance of their system and to understand how design features and process characteristics may impede or facilitate the implementation of IAEA safeguards, and it could be used for the selection of competing nuclear energy options for research, development and deployment.

However, further work is still required to consider whether the set of variables for the evaluation of indicators is complete; to examine the relationship between variables and indicators; to develop a rationale for the quantification of acceptance limits; and for the presentation of results.

Keywords: innovation, fuel cycles, IAEA, INPRO, methodology, proliferation resistance, basic principles, user requirements

1. INTRODUCTION

Existing scenarios for global energy use project that demand will at least double over the next 50 years. Electricity demand is projected to grow even faster. These scenarios suggest that the use of all available generating options, including nuclear energy, will inevitably be required to meet those demands [1].

In order for nuclear energy to play a meaningful role in the global energy supply in the foreseeable future, innovative approaches will be necessary to address concerns about economic competitiveness, environmental impact, safety, waste management, potential proliferation risks and necessary infrastructure. Considering these requirements and the future scenarios, the IAEA initiated the International Project on Innovative Nuclear Reactors and Fuel Cycles, referred to as INPRO, following resolutions of the IAEA General Conference in 2000.

The overall objectives of INPRO are:

- To help to ensure that nuclear energy is available to contribute in fulfilling, in a sustainable manner, the energy needs in the 21st century.

- To bring together all interested Member States, both technology holders and technology users, to consider jointly the international and national actions required to achieve desired innovations in nuclear reactors and fuel cycles that use sound and economically competitive technology, and that are based – to the extent possible – on systems with inherent safety features and minimize the risk of proliferation and the impact on the environment.
- To create a process involving all relevant stakeholders that will have an impact on, draw from, and complement the activities of existing institutions, as well as ongoing initiatives at the national and international level.

In order to fulfil these objectives, the first step of the project (Phase 1A) was dedicated to the definition of requirements, called basic principles, user requirements and criteria, that innovative nuclear energy systems (INS) should meet in several subject areas (economics, safety of nuclear installations, environment, waste management, proliferation resistance and infrastructure). A method to assess innovative nuclear energy systems on a national, regional or global basis, referred to as INPRO methodology, was developed. The result of Phase 1A was documented in the IAEA report: IAEA-TECDOC-1362, Guidance for the evaluation of innovative nuclear reactors and fuel cycles [2].

In the following step of the project (called Phase 1B, first part), validation and improvement of the INPRO methodology through several case studies were carried out, as well as further development of analytical tools. The results of this phase were documented in the IAEA report: IAEA-TECDOC 1434, Methodology for the assessment of innovative nuclear reactors and fuel cycles [3].

The INPRO results in the area of proliferation resistance are largely based on the consensus reached in October 2002 at a meeting held in Como, Italy [4], and at follow up meetings held in March 2004 in Cheju, Republic of Korea, and in September 2004 in Vienna, Austria, where the feedback from the case studies was taken into account.

2. PROLIFERATION RESISTANCE

In designing future nuclear energy systems, it is important to consider the potential for such systems to be misused for the purpose of producing nuclear weapons. This is one of the key objectives of the international non-proliferation regime, with its many national and multinational agreements and institutions; the IAEA safeguards system is a fundamental element of this regime. While almost any INS can be adequately safeguarded with sufficient effort and resources, the cost of providing safeguards assurances depends on the nature of the nuclear fuel cycles of a State. Should nuclear power based on existing technologies greatly expand, detecting any diversion of civilian nuclear material or the misuse of facilities dedicated to the peaceful use of nuclear energy, or undeclared nuclear materials or activities, will become increasingly costly.

Therefore, in designing future nuclear systems it is essential to increase their proliferation resistance. In this context *Proliferation Resistance* refers to the characteristics of a nuclear energy system that impede the diversion or undeclared production of nuclear material, or misuse of technology, by States intent on acquiring nuclear weapons or other nuclear explosive devices [4].

The degree of proliferation resistance results from a combination of technical design features, operational modalities, institutional arrangements and safeguards measures. These can be classified as *intrinsic features* and *extrinsic measures*. Intrinsic features result from the technical design of INS including those that facilitate the implementation of extrinsic measures. Extrinsic measures are based on States' decisions and undertakings related to nuclear energy systems.

Intrinsic features consist of technical features that:

a) reduce the attractiveness for nuclear weapons programmes of nuclear material during production, use, transport, storage and disposal, including material characteristics such as isotopic content, chemical form, bulk and mass, and radiation properties;
b) prevent or inhibit the diversion of nuclear material, including the confining of nuclear material to locations with limited points of access, and materials that are difficult to move without being detected because of size, weight, or radiation;

c) prevent or inhibit the undeclared production of direct-use material, including reactors designed to prevent undeclared target materials from being irradiated in or near the core of a reactor; reactor cores with small reactivity margins that would prevent operation of the reactor with undeclared targets; and fuel cycle facilities and processes that are difficult to modify; and

d) facilitate nuclear material accounting and verification, including continuity of knowledge.

Five categories of extrinsic features are defined, as follows:

a) commitments, obligations and policies of States, such as the Treaty on the Non-Proliferation of Nuclear Weapons and the IAEA safeguards agreements and protocols additional to such agreements;

b) agreements between exporting and importing states on exclusive use of nuclear energy systems for agreed purposes;

c) commercial, legal or institutional arrangements that control access to nuclear material and technology;

d) verification measures by the IAEA or by regional, bilateral or national measures; and

e) legal and institutional measures to address violations of measures defined above.

The assessment of proliferation resistance by the INPRO assessment methodology is limited to the proliferation by States and does not include protection against the theft of fissile material by sub-national groups.

3. INPRO Assessment of Proliferation Resistance

INPRO aims at assessing the whole INS in a specific State or region, not only separate elements of an INS. For the assessment of the proliferation resistance of an INS INPRO has established *basic principles* and *user requirements* that an INS is expected to fulfill. To evaluate the fulfillment of user requirements, for each user requirement *indicators* and *acceptance limits* have been defined. The detailed nature of indicators may be described by *variables.*

The assessment is done primarily bottom-up, starting with the evaluation of indicators. The indicators and their acceptance limits then are used to evaluate the fulfillment of user requirements and the compliance with the basic principles. Adopting the Agency's concept of an integrated safeguards approach for a State as a whole, in addition a top-down element has been introduced by the analysis of plausible acquisition paths to be covered by *multiple proliferation features and measures.*

3.1 Basic Principles and User Requirements

Two basic principles have been defined that provide high-level guidance regarding innovative nuclear energy systems:

1. *Proliferation resistant features and measures shall be implemented throughout the full life cycle for INS to help ensure that INS will continue to be an unattractive means to acquire fissile material for a nuclear weapons programme.*

2. *Both intrinsic features and extrinsic measures are essential, and neither should be considered sufficient by itself.*

These basic principles emphasize the importance of both intrinsic features and extrinsic measures. Regardless of the effectiveness of intrinsic features, extrinsic measures will always be required. Another important aspect is that intrinsic features and extrinsic measures for proliferation resistance are to be implemented through the full life cycle of an innovative nuclear system, and that the complete INS will be assessed for proliferation resistance and not only individual elements of a fuel cycle.

For each basic principle top-level user requirements have been defined to provide guidance for designers and decision makers how intrinsic features and extrinsic measures can be implemented. For the first basic principle three top-level user requirements have been established:

1. *States' commitments, obligations and policies regarding non-proliferation should be adequate.*

2. *The attractiveness of nuclear material in an INS for a nuclear weapons programme should be low. This includes the attractiveness of undeclared nuclear material that could credibly be produced or processed in the INS.*

3. *The diversion of nuclear material should be reasonably difficult and detectable. Diversion includes the use of an INS facility for the introduction, production or processing of undeclared nuclear material.*

States' commitments, obligations and policies regarding non-proliferation as extrinsic measures have considerable impact on the proliferation resistance of an INS. Another important factor is the attractiveness of the nuclear material. The attractiveness of nuclear material reflects its suitability for conversion into nuclear devices. It refers, however, also to processes by which weapons usable material could be acquired. Less attractive nuclear material may have also an impact on the safeguards effort to be spent by the IAEA.

The key barrier to proliferation is the difficulty of diversion and the risk of detection. Difficulty of diversion and its detectability should be high enough to make the implementation of an independent nuclear weapons program more attractive than the use of the INS. Difficulty and detectability may be also summarized as *safeguardability* meaning in this context a feature of an INS that impedes or facilitates the implementation of IAEA safeguards. Safeguardability depends on the material, on the processes and on design characteristics of the elements of an INS.

For the second basic principle two top-level user requirements have been established:

1. *Innovative nuclear energy systems should incorporate multiple proliferation resistance features and measures.*

2. *The combination of intrinsic features and extrinsic measures, compatible with other design considerations, should be optimized (in the design/engineering phase) to provide cost efficient proliferation resistance.*

It is understood that each plausible acquisition path is covered by appropriate verification measures, and that plausible acquisition paths should be covered additionally by multiple intrinsic features. However, one also has to take into account that some intrinsic features may adversely affect the implementation of IAEA safeguards.

Every INS, in principle, can be adequately safeguarded, provided sufficient resources are available, but the effort required to implement verification measures varies. During development of an INS, intrinsic design features that will reduce the implementation costs of IAEA safeguards should be considered. It is recognized that there are cost trade-offs between intrinsic features and extrinsic measures, and their optimization for cost effectiveness is encouraged. It is, however, also recognized that the features and measures must be compatible with other design considerations such as safety and economics, and that the costs for the implementation of IAEA safeguards must be reasonable.

3.2 Indicators and variables for the evaluation

For each user requirement *indicators* and *acceptance limits* have been defined to assess their fulfillment. The indicators describe how user requirements are to be fulfilled. Acceptance limits describe the minimum fulfillment required for each indicator.

For the evaluation of indicators furthermore a set of *variables* has been established which requires, however, still further work. The variables are basically intrinsic features and extrinsic measures describing States' commitments, obligations and policies regarding non-proliferation, material

characteristics such as isotopic content, chemical form, radiation field or heat generated, and facility and process characteristics such as design features that limit access to nuclear material, material stock and flows or time required to divert or produce nuclear material and convert it to weapons useable form. The variables are, in principal, similar to the material, technical and institutional barriers, and their attributes as given in the TOPS report [5,6].

3.3 Assessment of proliferation resistance

Application of the user requirements, indicators and acceptance limits requires an accepted means to assess the proliferation resistance of a nuclear energy system, using clear and transparent tools. Such assessments would serve a number of diverse uses. It would help designers to assess the impact of intrinsic design features on the overall proliferation resistance of their system and to understand how design features and process characteristics may impede or facilitate the implementation of IAEA safeguards. This would allow providing cost efficient proliferation resistance balancing intrinsic features and extrinsic measures. Finally, an assessment method could be used in making such decisions as the selection of competing nuclear energy options for research, development and deployment.

It is widely recognized that a common assessment method needs to be developed that will allow such determinations to be made in a consistent manner. The proliferation resistance assessment method will likely be a composite incorporating scenario-based and attribute-based tools similar to those used in safety and physical security analysis to examine material and facility targets within an INS [7].

4. Further Development

The Terms of Reference for Phase 1B and Phase 2, confirmed by the 7th Steering Committee, contain the continuous improvement of INPRO Methodology with a focus on a more quantitative approach as an activity to be implemented during the Phase 1B.

In this context, in addition to five other assessments of INSs by various States using the updated INPRO methodology, the Republic of Korea agreed to perform a further case study, looking at the whole fuel cycle of DUPIC (direct use of the PWR spent fuel in CANDU reactors), in order to:

- assess the proliferation resistance characteristics of the DUPIC fuel cycle using the revised INPRO methodology, and
- develop recommendations on the further improvement of the INPRO methodology in the area of proliferation resistance for its application to evaluation of innovative nuclear energy systems.

The primary goal, at this stage, is the further development of the assessment methodology and its components: completion of the list of variables, examination of the relationship between variables and indicators, quantification of variables and indicators and the rational for such quantification, and the development of a model for the aggregation and presentation of evaluation results to designers and responsible authorities.

The results of the case study will then be used as input to the Joint Study on the assessment of an innovative nuclear energy system based on a closed nuclear fuel cycle with fast reactors using the INPRO methodology, which would allow a wider scope of the proliferation resistance assessment. Partners for this study are China, France, India, the Republic of Korea and the Russian Federation, with Japan as an observer.

Additionally INPRO seeks to enhance the collaboration, on a complementary and synergetic basis, with other national and international initiatives, e.g. Generation IV International Forum (GIF).

References

[1] INTERNATIONAL PANEL ON CLIMATE CHANGE, Special Report on Emission Scenarios, A Special Report of Working Group III, Cambridge University Press, Cambridge (2000) (<http://www.grida.no/climate/ipcc/emission/index.htm>)

[2] INTERNATIONAL ATOMIC ENERGY AGENCY; Guidance for the evaluation of innovative nuclear reactors and fuel cycles, Report of Phase 1A of the International Project on Innovative Nuclear Reactors and Fuel Cycles, TECDOC-1362, IAEA, Vienna (2003).

[3] INTERNATIONAL ATOMIC ENERGY AGENCY; Methodology for the assessment of innovative nuclear reactors and fuel cycles, Report of Phase 1B (first part) of the International Project on Innovative Nuclear Reactors and Fuel Cycles, TECDOC-1434, IAEA, Vienna (2004).

[4] INTERNATIONAL ATOMIC ENERGY AGENCY, DEPARTMENT OF SAFEGUARDS; Proliferation Resistance Fundamentals for Future Nuclear Energy Systems, STR-332, IAEA, Vienna (2002).

[5] UNITED STATES DEPARTMENT OF ENERGY, NUCLEAR ENERGY RESEARCH ADVISORY GROUP, Technological Opportunities To Increase the Proliferation Resistance of Global Civilian Nuclear Power Systems (TOPS), Report by the TOPS Task Force of the Nuclear Energy Research Advisory Committee (NERAC). January 2001.

[6] UNITED STATES DEPARTMENT OF ENERGY, NUCLEAR ENERGY RESEARCH ADVISORY GROUP, Annex, Attributes of Proliferation Resistance for Civilian Nuclear Power Systems, NERAC, October 2000.

[7] PACIFIC NORTHWEST NATIONAL LABORATORY, Guidelines for the Performance of Non-proliferation Assessment, PNNL-14294, May 2003.

Session 9

Modelling and Data Treatment

Monitoring of Nuclear Material Movements at a Reprocessing Plant

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Abstract:

A process monitoring software tool has been developed, which allows inspectors a remote verification of nuclear material movements based on the interpretation of neutron and gamma signals, which are measured during the trajectory and online analysed. Neither supervision nor control is intended with the software but a checking of the required coherency and conformity with safeguards purposes. The data analysis and interpretation kernel supports the inspector in the back-end of the nuclear fuel cycle and allows in particular near real time accountancy in a reprocessing plant.

To monitor the nuclear material flow for near real time accountancy purposes a diagnostic aid for the inspectorate was conceived, which interprets data from solution and powder measurements of different kind and at different steps in the process.

In a first step tank transfers have been analysed by syntactic pattern recognition and introduction of functional blocks. A predefined sequence of functional blocks yields a tank signature and characterises unambiguously the filling rates, normal operation tank levels, levels for stirring and skimming, and the emptying rates. In a second step the same approach has been applied to the cyclic behaviour of a weighing scale. The weighing procedure with first scale-recalibration check, with the filling of Pu powder cans and with the final inventory weight determination could be verified remotely. Finally this syntactic pattern recognition approach has been applied to follow the movement of nuclear material by treating the neutron and gamma signals.

This paper reports on the analysis of the neutron and gamma signals for the remote follow-up of the spent fuel assemblies on their way to and from the measurement pit. A real-time analysis of the signal from the neutron counter simultaneously to the signal of the gamma counter positioned under a fixed geometry allows an identification of a movement signature. The functional blocks describing the movement signature consist of peaks and plateau's. The auto-correlation of a predefined sequence with the real sequence based on the measured neutron and gamma counts allows to characterise the nuclear material moved in the channel towards and from the measurement pit. With the appropriate parameters such as peak height and plateau length the software monitors if the material movement procedure is followed conform to the safeguards purposes.

Keywords: process monitoring; neutron/gamma signal interpretation.

1. Introduction

Process monitoring is an excellent tool to verify absences of undeclared activities or anomalities in industrial or semi-industrial processes. A software tool has been developed upon request of the Euratom inspectorate to monitor in real time the nuclear material (NM) flow of about twice 850 tons/yr through the whole PUREX process in UP2 and UP3 for the spent fuel of power reactors at the La Hague site. The about 800 kg of Pu

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processed per month through each line are for safeguards purposes measured accurately (with an error less than 1%, corresponding to less than 8 kg per month). However it is not satisfying to verify just the interim inventory, because it only identifies an eventual problem without any localisation in place and time and without any indication of the reason. Therefore it is needed to monitor the complete material flow, while surveying if the spent fuel batch is processed following the declared flow sheets and complying with the safeguards requirements on material losses. The monitoring includes:

- the status of health check
- the surveillance through gamma/ neutron counters for the displacement of nuclear material
- the NM liquid inventory follow-up through level and density measurements for the transfers between tanks,
- the NM powder inventory follow-up through the weighing scale signal for the final filling of recipients with the Pu powder product.

It especially verifies for Near Real Time Accountancy purposes the coherency with safeguards purposes and conformity in procedure.

2. Structure of the monitoring software

2.1. Data acquisition with a data historian

Process industries commonly work with a superposition of different acquisition systems. The data over a large time period, are usually collected in a first time-series database of non-relational type and compressed by a data historian, that allows fast real-time access to the significant reported data. The reported data are analysed and the results are exported in a second database of relational type.

Each significant signal variation has to be unambiguously detected with time and date stamp. The data historian either filters out data points at relatively large time intervals in case of no or constant variation of the signal (e.g. the auto regressive moving average filter) or the historian reports exceptions (exceeding the specified compression deviation blanket).

The software developed for solution monitoring was not intended to duplicate the operation of a data historian and therefore neither serves to compress nor to filter the original data but utilises the commercially available data historians such as Osisoft PI [1] or the Matrikon OPC [2].

2.2. Monitoring of a batch process by the Data Analysis and Interpretation (DAI) software

2.2.1 Data analysis & interpretation concept

As the nuclear material of the spent fuel is processed in batch mode or even in continuous mode, a certain cyclic behaviour in the measured signal profile can be observed. However the profiles of the measured signals in this large industrial plant is not strictly periodical and therefore no standard pattern recognition algorithms can be applied as such.

The innovative algorithm in the software is to detect the predefined elementary functional behaviours in the signal. These functional behaviours represent in a small block one function in the process cycle. The process cycle can be defined as a sequence of events with certain functions. The specific task of the software is to interpret the recorded points of the data historian and to detect the different events by recognising the sequential functional behaviours with an accurate identification of the start and end of those functional behaviours.

The innovative aspect is that the data interpretation is done on the following abstract level:

- each typical process cycle is modelled by composing series of succeeding functional behaviour blocks, and
- the analysis is completely performed in a symbolic way by evaluating the sequence of functional behaviour blocks.

As the NM monitoring has to operate efficiently, the analysis includes also all basic information of the plant design and of the procedure for each reprocessing step. The functional behaviour, that is associated to the measured signal, is online compared with the possible functional behaviours, that are predefined in the plant design.

To analyse the evolution of the signal in time, an observation window moves from one data point to the next and calculates for each point the average of the signal variation over the time with the data point values, just before and after the window. This characterises each data point with a mean variation, which allows the detection of the start of a new functional behaviour by comparing with the predefined variations with lower and upper limits. A detailed description is given in [3].

2.2.2 Cross-correlation

To facilitate finding out the correlated signals a status code is introduced to associated functional behaviour with the following rules:

- all potentially correlated signals are marked by a change in status code of the associated functional behaviour from passive to ready
- a significant signal variation for a ready functional behaviour changes its status into active if no additional requirements have to be fulfilled
- a significant signal variation for a ready functional behaviour changes its status into checking in case of additional requirements and passes finally to active if all requirements are fulfilled.

The DAI kernel evaluates all correlated functional behaviours, which are simultaneously in an active mode. By means of cross-correlation the comprehensiveness between the active correlated functional behaviours is checked and coherency with safeguards purposes verified.

2.2.3 Auto-correlation

The repetition of a certain process cycle is recognised as the looping of a sequence of functional behaviours. This looping is controlled by means of auto-correlation, this means that the completeness of a cycle is verified by comparing with the predefined design. Errors, such as incompleteness of the cycle, inadvertent repetition of a functional behaviour, or falling out of a part of the process cycle, are reported with a diagnosis on the global constraint violation.

3. DAI application on the neutron & gamma signals

3.1 The fuel insertion process step

All the spent fuel (SF) assemblies with their declared characterisation are at receipt roughly checked by total weighing and passive neutron counting before storing in the spent fuel pool. Before entering the process a second more accurate check is foreseen with the burn-up detector. The first process step can be localised in the input cell for the fuel insertion with the burn-up check. Thereby the fuel is taken from the pool and moved inwards the measuring pit, moved backwards and tilted for introducing in the chopping channel. To follow the fuel movement in the input cell a neutron counter and gamma spectrometer are placed at the entrance of the chopping channel. The different movements of the spent fuel assembly in the input cell are presented in Fig.1. (a-b-c-d-e-f-g-h).

With the data historian the neutron and gamma signals are collected and compressed. A homothetic scaling is applied to standardise the signal range and to feed the DAI software with clear cyclic signals.

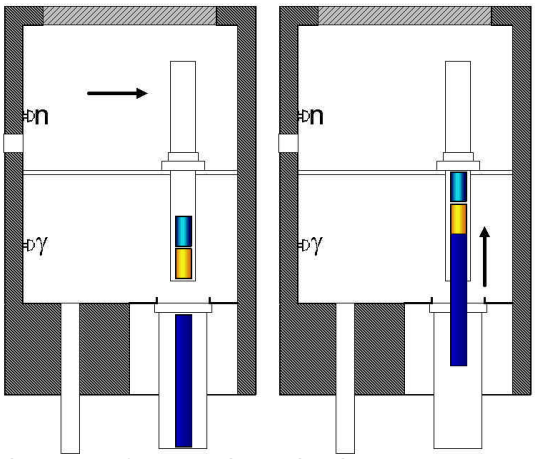

Fig.1a: SF from pool Fig.1b: SF transport

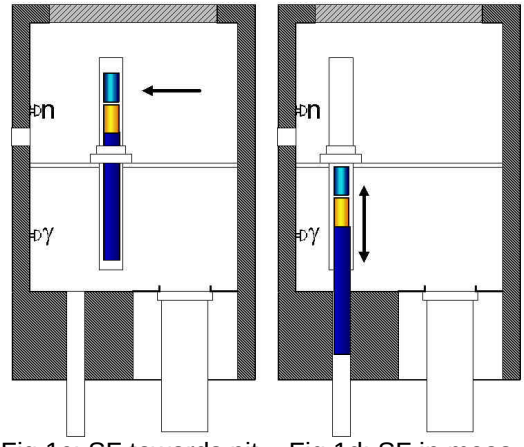

Fig.1c: SF towards pit Fig.1d: SF in meas. pit

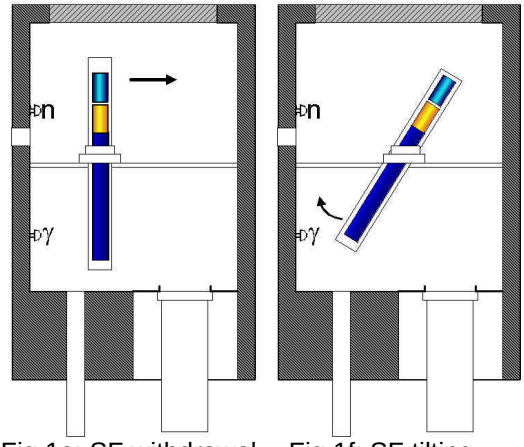

Fig.1e: SF withdrawal Fig.1f: SF tilting

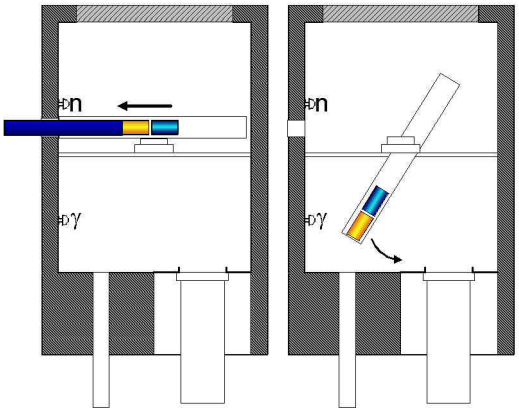

Fig.1g: insert chopper Fig.1h: reset

3.2 The neutron/gamma cyclic signals in the input cell

Both signals, gamma and neutron that are reported by the data historian are analysed and interpreted simultaneously by DAI.
Two typical cycles, rescaled by the data historian, are presented in Fig. 2. The neutron and gamma signal behave in a similar way, but the presence of both with the given scale are characterising the real spent fuel composition. During the SF movement from the SF storage pool to the measurement pit, the SF approaches differently the detectors. As soon as the fuel is entering the solid angles of respectively the gamma and neutron detectors, the count rate directly goes up to a certain plateau, which is specific for the isotopic content of the spent fuel pellets in the Zirkaloy cladding. The waiting of the fuel in the loading machine before the start of the movement does not alter the registration in the solid angle of the detectors and thus gives an almost constant count rate. As soon as the fuel is approaching the detector by horizontal displacement a steep increase in signal is observed.
Once the fuel is descending in the measurement pit a very fast change in count rate is observed. The head and end of the spent fuel pins are of stainless steel and under the long-term irradiation Co-60 isotopes are present. The passing by of the head/end of a fuel leads to significant increases in count rate of the detectors and explains the first peak of Fig. 2. Once the fuel is completely in the measurement pit, zero counts are recorded. The number of counts increases again very fast when the fuel is moved out of the measurement pit, especially a second peak is recorded when the head/end is passing by the detectors. Then the fuel is tilted from its vertical position to a horizontal position in order to enter the chopping channel. At about 45° tilted position the head/end of fuel is passing closely at the gamma counter and this is recognised by the third peak in Fig. 2. Finally the fuel is inserted in the chopping channel and this is registered with the second plateau, at another level than the first one because the detectors have a different view on the fuel in the horizontal position.

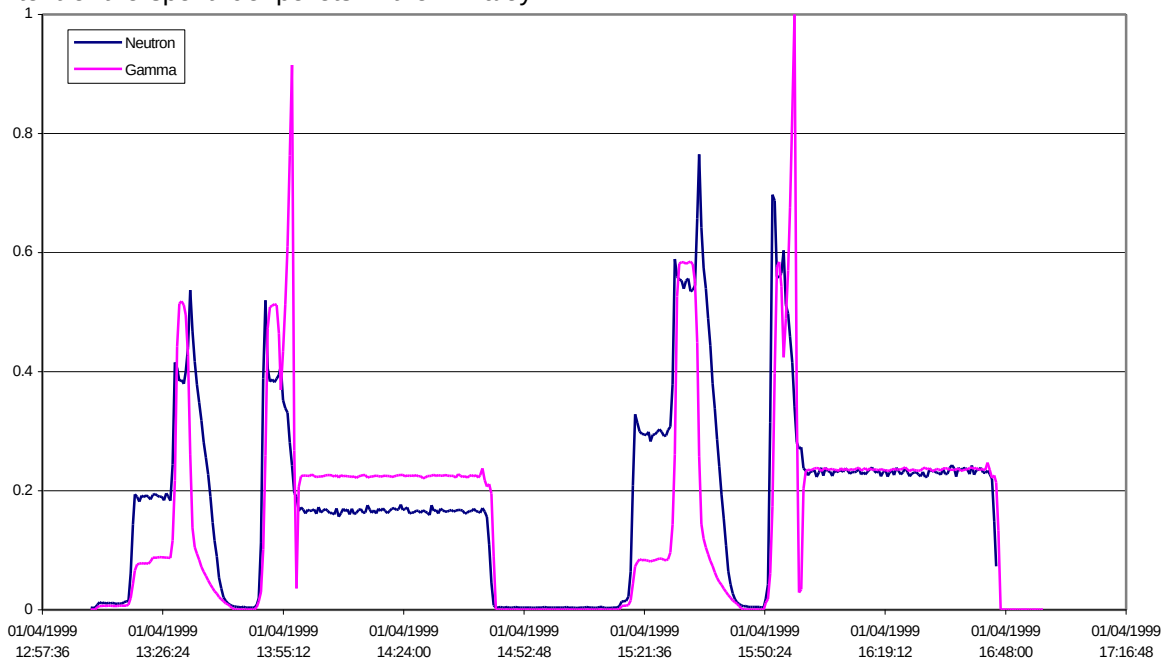

Fig. 2: Scaled n/gamma signal typically returned by the data historian for two consecutive cyclic input operations on the SF.

3.3 Symbolic modeling with the DAI software for automatic recognition of the neutron/gamma cycles

A fuel in the loading machine waiting an OK for movement can be identified with a plateau whereas a horizontal or vertical displacement or even a tilting of the spent fuel assembly is characterised with a steep slope. This is modelled with three normal kinds of functional behaviour:

- increasing slope with unknown value,
- decreasing slope with unknown value,
- plateau with unknown value.

For the detailed definition of these functional behaviours the reader is referred to the DAI user manual [4].

To analyse the signal for one cycle three subregions are indicated:

The first part concerns the displacement of the fuel in vertical position; The second part concerns the tilting of the fuel and the third part covers the insertion of the fuel into the chopping channel.

The symbolic model for one loop of the cyclic gamma/neutron signal is given in Fig. 3. A waiting time after the tilting is allowed in the model by the optional “plateau after decreasing slope” functional behaviour that can be bypassed. A synchronisation point has been introduced in the model after the insertion in the chopping channel. This has been selected by the modeller on a free basis at the end of the cycle to anticipate difficulties in addressing the complete loop and to allow resetting of the analysis for the next loop.

The fine tuning of all parameters for all functional behaviours was done with all available neutron /gamma signals, in particular the deviations of the standard signal, as shown in Figs. 4 and 5.

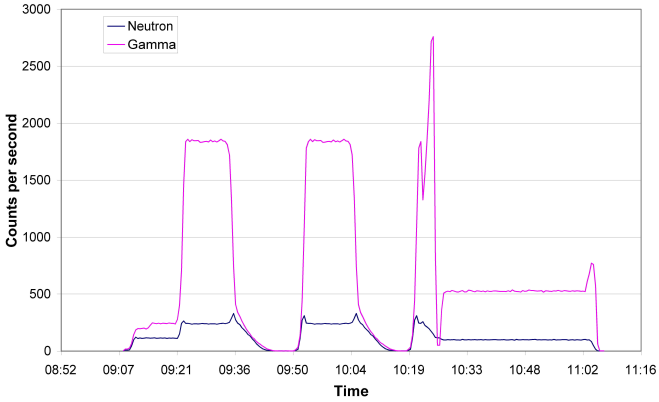

Fig. 4: Neutron and gamma signal for first case of common anomality: repeated (two) measurements in the pit before the approval for insertion in the chopping channel

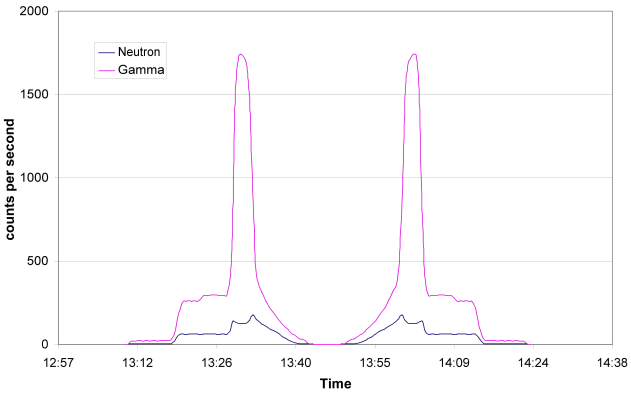

Fig. 5: Neutron and gamma signal for second type of anomaly with absence of the second part, because the spent fuel is only moved from pool to measurement and back and is not inserted in the chopping channel.

3.4 Final application: inspector tool for automatic follow-up of the spent fuel in the input cell.

This application example of DAI on the neutron-gamma signal allows the inspector to monitor the nuclear material of the SF into the chopping channel and after a more detailed characterisation of the SF with the burnup detector in the measurement pit. In view of the Near Real Time Accountancy a diagnostic support for the inspectors is in direct use with this DAI software. This application illustrates how the DAI software can utilise the described algorithms of cross- and auto-correlation and the formalisms of symbolising the signal profile and of assigning a status to an event in a generic way.

The displacement of the fuel from the storage pool into the chopping channel is correlated with the burnup characterisation of this SF assembly. The coherence in nuclear material flow further down in the process with the burnup signature can be verified in real time and each error is alarmed with a diagnosis (inclusive the necessary messages and early warnings). Moreover, the conformity of the consecutive actions in the fuel insertion process step with the prescribed model is evaluated in an efficient and transparent way.

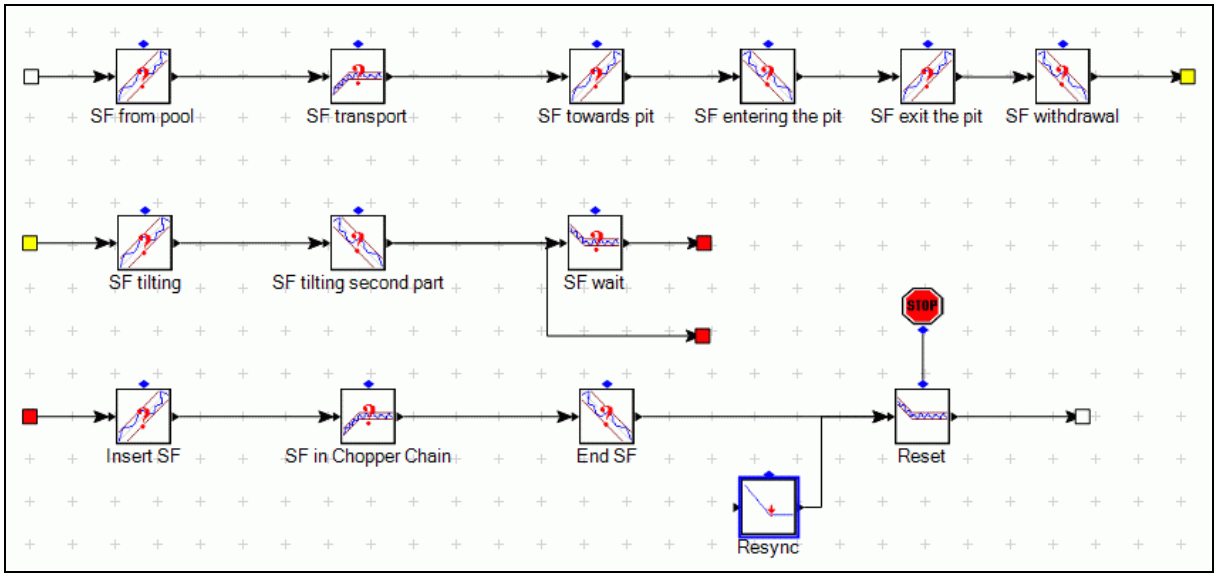

Fig. 3: Model for the cyclic neutron/gamma signal

Thanks to the pragmatic use of all on-line experience with the process, especially the vast amount of neutron/gamma signals, the monitoring system is tailored with a minimum of disabilities in the recognition of a cycle and a maximum of transparency. Those anomalies, which are no longer transparent, are due to the need of a higher level diagnosis, which is not envisaged with this process monitoring tool. The appearance of these anomalies is rare and neglecting them is acceptable according to Howell [5].
With the other applications, on the tank level/density for solution transfers as reported in [6], and on the weighing scale for the powder quantification as reported in [7], this application also demonstrates the general concept and modular structure of the DAI programme, that allows the analysis of very different kind of signals.

4. Conclusion

4.1 Results

A stand-alone software tool, that monitors the nuclear material flow for the different process steps, is successfully implemented at UP2 and UP3 facilities of the La Hague Reprocessing Plant and in operation under the so-called “System 7” application by the Euratom inspectorate. In a reprocessing plant the signals for each process step, such as the spent fuel insertion process step as addressed in detail here, are not only interpreted with identification of all functional behaviours but also evaluated carefully concerning the integrity of the material flow from and to the neighbouring places. Both coherence in material flow and conformity with the prescribed safeguards procedure are analysed and for each error an alarm and a diagnosis is given. A precise follow-up of the functional behaviour and of the sequence of behaviours allows a precise accounting of the mass of nuclear materials at the key measurement point in a material balance area.
The tool serves the inspector as a verification means for coupling the displace spent fuel assembly with its burnup signature. At each point, where an anomaly is detected, the cycle is interrupted. The tool then lists all possible procedural errors or constraint violations. It is left to the inspector to decide on the real cause, to remain consistent with their defined responsibility in the follow-up of the process. The tool is setup in a generic way, so that it can be applied for very different type of signals, such as the isotopic at different sites. Moreover this example demonstrates the usefulness of a simplified approach to follow-up the complete transfer of different material flows, in which all functional blocks of the design have been successfully validated.

4.2 Perspectives

The EURATOM inspectorate has asked to implement a similar monitoring tool at the reprocessing plant of Sellafield (THORP). The software has also been selected by the IAEA for the solution monitoring of the TETRA

demonstration plant by the Japanese inspectors of NMCC, and for the inspection of the Tokai reprocessing plant (TRP) by IAEA itself. Apart from safeguards inspections the tool can be applied as a batch norm monitoring system (ISA-88 and ISA-95). The batch industry is still finalising the last ISA-95 norm to enhance the Material Flow Management and the traceability in Manufacturing Executing System (MES).

References

[1] Osisoft PI: user manual and documentation via web: http://www.osisoft.com/5_267.aspx

[2] Matrikon OPC: user manual and documentation via web of Matrikon: http://www.matrikonopc.dom/index.html

[3] Janssens-Maenhout, G. and Dechamp, L, Process Monitoring for Near Real Time Accountancy, *J. Inst. Nucl. Mat. Man.* XXXII (4), Spring 2004

[4] Janssens-Maenhout, G. and Dechamp, L. *DAI User manual*, JRC special publication, January 2005

[5] Scothern, S.J. and Howell, J. 1997. A Physical-Model-Based diagnostic aid for Safeguarding Nuclear Material in a Liquor Storage Facility, *J. Inst. Nucl. Mat. Man.*, XXV(4), 20-29.

[6] Thevenon, J.B., Janssens-Maenhout, G., Dransart, P. and Dechamp, L., Une approche du monitoring d'installation industrielle: exemple d'une application EURATOM pour COGEMA à La Hague, *Clubautomation,* Oct. 2003.

[7] Janssens-Maenhout, G. and Dechamp, L. and Brian, H. Process Monitoring at a Reprocessing Plant : Example of a Weighing Cycle, *Proc. of the ESARDA Ann. Conf. (Stockholm)*, May, 2003

Radiation Analysis for Monitoring the Presence of Nuclear Material

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Abstract:

This paper describes an experimental study of the use of radiation patterns in a multi detector environment to identify the position of nuclear material. The experimental configuration uses neutron detectors and the methodology is based on pattern matching. The paper provides experimental results and assesses the feasibility of radiation-based localization of nuclear material samples. The testing was carried out in the JRC PERLA laboratory. In addition the paper describes software tools for validation of the quality of the experimental data being used to test the method. These include off-line visualization and analysis of the neutron time series to verify that the measurements have been made using experimental data that is a stationary process.

Keywords: nuclear material monitoring, pattern matching, signal monitoring, sequential testing.

1. Introduction

For both security and safeguards, continuous monitoring of the location and movement of nuclear material is of importance in storage areas and in workshops. Monitoring systems for material in storage or moving along very restricted pathways have been implemented by means of radio tags, heat detectors, vision systems and radiation detection. The data is interpreted to confirm either that movement is not taking place (storage) or that a movement of a standard type has taken place. In the latter case, the interpretation is aided by the very restricted pathway envisaged. The radiation signal confirms the presence of NM of the reference type and the evolution of the signal through time in a sequence of detectors along the pathway illustrates the fact of movement. The use being made of the radiation data is not based on an estimation of position of the source. The use of radiation data to characterize position in some general way would allow remote inspection systems to contribute to reliable monitoring in a wider variety of situations. A number of experimental studies have examined the possibility of using radiation monitoring to determine the position of a source without any a priori hypothesis about its position or pathway. In these experiments, the source is somewhere in a two dimensional space where some freedom of position or movement are foreseen.

In /1/ and /2/ the authors are motivated by the detection of neutron sources in reactor diagnostics using a very small detector which can fit in the instrumentation tube of a PWR. For this reason they study the situation of a source in a tank of water. The experiments study the location of the source in a range of 1 to 10 cm from the detector. In this isotropic experiment the neutron flux and its gradient show useful regularity as a function of distance (≤ 10cms) and for this range an estimate of distance can be made having error in the order of 8%.

In /3/ the authors describe the use of three scintillator detectors to locate a sample inside a circle of 100 cm. The source position can be determined with an error of less than one cm. In this experiment however the measurement time is 12 hours for the localization of a single position. Again the success of the method is also helped by the isotropic nature of the experiment.

In /4/ the authors study a source in a room of dimension 30x50 ft. The method uses adaptive estimation to track the movement of the source using 4 gamma detectors. The methodology of /4/ assumes that at each time point the background is the same at each detector. In the experiments the source was moved along a looped pathway defined by 17 points with a count time of 3 minutes at each point. The results indicated that the source could be tracked with an error between one and two feet. Again the analysis method was for a space without furniture (i.e. isotropic).

A robust method must be able to deal with large spaces having variable background at different detectors and with furniture and other moderating material throughout the space thereby rendering the radiation absorption and transport non-isotropic. These conditions are characteristic of the PERLA laboratory for example where three stores provide a considerable gradient in background.

2. The PERLA laboratory

PERLA is the EU Commission NDA calibration and training laboratory which contains a number of experimental work zones each having a variety of equipment. Sealed samples of fissile material and sources are used in work aimed at supporting the implementation of Euratom and IAEA safeguards /5/. PERLA provides a ready-made infrastructure for experiments concerned with location of a sample of Pu bearing material. The laboratory area (approx 20m by 10m) is monitored by neutron detectors (^{3}He proportional counters) attached to the wall in ten positions around the circumference of the laboratory. Each position has a pair of detectors (see Figure 1). One of each pair is enclosed in 2 cm of polyethylene moderator with a cadmium sleeve to detect only fast neutrons and the second is a naked detector to detect thermal neutrons. The system uses 60 cm ^{3}He tubes with a high voltage of 1,5 kV. The acquisition is done with a PC on a NI-Lab-PC-120AI board. The detector system was installed to monitor the background conditions in PERLA for quality control purposes. This acquisition system was produced by TESYS Technology & Systems (http://users.libero.it/tesys) and the counting software was programmed by Syrea srl. (http://www.syrea.it). These detectors continuously measure the neutron count rate from the fissile material and sources in the laboratory (counting interval 1 minute). The data files from this system are fed into a customized software system to be used for the localization experiments described here. The results given here are based on the signals from the moderated detectors only. While the general approach can be used for a 3-D localization, the results reported here are of a 2-D localization experiment.

Figure 1: one of the 10 detector stations

In a situation without movement in which a specific sample is in a specific location, the neutron count rate at any detector will depend on the neutron emission rate of the sample, the distance between the sample and the detector and on the relative locations of furniture and equipment that will affect the neutron transport and absorption. Each location for a neutron emitting sample is associated with a vector of count rates (here 'vector' refers to the ten count rates detected by the 10 detectors). These gross count rates will also depend on the background signal that would be present if the sample were absent. The purpose of the experiment is to test the hypothesis that the position of the sample could be deduced from the vector of net count rates. The method is basically to create a calibration archive consisting of the net count rate vectors corresponding to the variety of positions within the laboratory. This is done experimentally. Once the archive has been established, the unknown location of a Pu sample in the laboratory is "found" by comparing the radiation vector of the

unknown location with the archive set of vectors until some match is found. The experiments described here have been made to test this method.

In order to limit the work of creating the calibration archive, the possible positions for the sample were limited to a grid of 36 points (defined in terms of the large tiles covering the floor). The positions of these points are shown in Figure 2 below which also shows the positions of the detectors. The experimental results reported in section 5, relate to "unknown" locations that are one of these grid points. The sample which was used to create the calibration archive (referred to later as ' L') was a PuO_2 sample containing circa 1000 g of Pu.

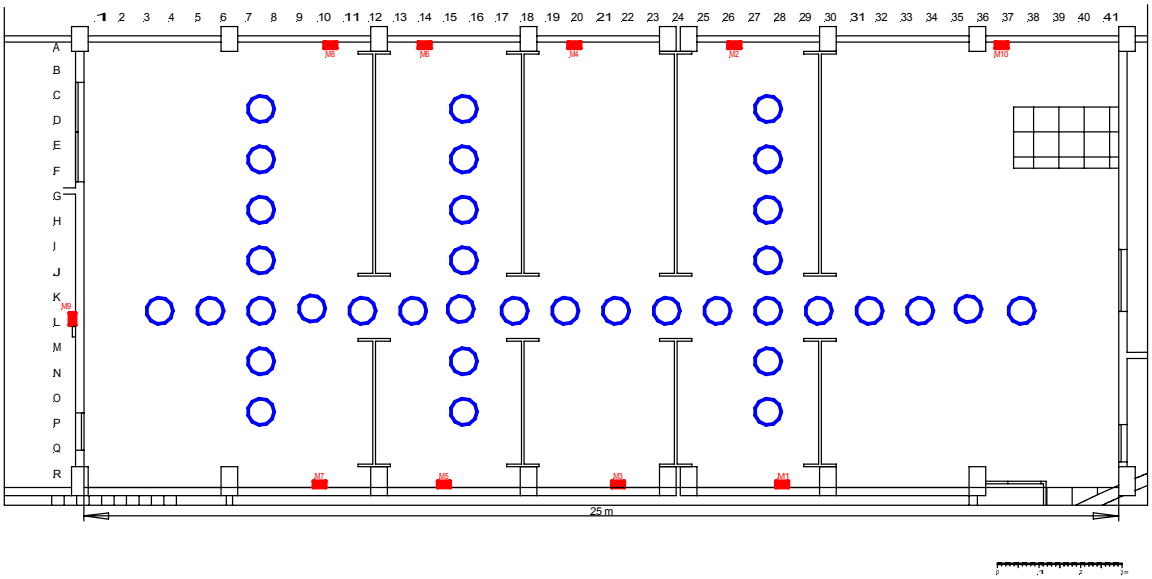

Figure 2: Calibration grid (blue) and detectors (red)

Note that the coordinate system step-length (integer or letter on Figure 2), represents steps of 50 cm. The vertical or horizontal distance between two neighbouring calibration points (blue) is 1 metre.

3. Recognising Locations by using Similarity of Rate Vectors

The rate vectors described earlier can be represented in the notation $R^{(u,v)}_{net,sample,i}$ where (u,v) denotes the co-ordinates of the position of the sample and 'i' denotes the identity of the detector. Earlier we spoke about the rate vector which would be a characteristic of the position (u, v). This characteristic rate vector is just the set of $R^{(u,v)}_{net,sample,i}$ for i=1,2,…**K** where **K** is the number of detectors. This vector of K values can be denoted $R^{(u,v)}_{net,sample}$. When the identity of the sample is unambiguous this vector will be denoted $R^{(u,v)}_{net}$. This set of values is the vector or "profile" of a particular sample in a particular position. Using a standard sample, we have created a vector $R^{(u,v)}_{net}$ for each grid position (u,v) in the experimental area. The database of these vectors is called the calibration archive for the method.

Once the archive has been created, we go on to consider a new profile created by a sample in an unknown position. This sample could be the same sample used to create the calibration archive or a different sample. We wish to "locate" the unknown position of the sample by searching in the calibration archive for a vector "matching" the vector of the unlocated sample. To do this we need to have a numerical concept of 'similarity of two profiles' that we now go on to define.

Computation of Similarity of Two Net Count Rate Vectors

In describing the computation of similarity index, we consider two vectors of net count rates that are denoted $R_{net}^{(1)}$ and $R_{net}^{(2)}$ where the superscripts (1) and (2) represent the positions ($\mathbf{u_1,v_1}$) and ($\mathbf{u_2,v_2}$) of the two samples. The computation of similarity is carried out on the vectors of components of $R_{net}^{(1)}$ and $R_{net}^{(2)}$. To simplify the appearance of the formulae however, the count rates will be represented in a simpler way in which,

$$x_i \quad \textbf{denotes} \quad R_{net,sample-1,i}^{(1)}$$

$$\textbf{and} \quad y_i \quad \textbf{denotes} \quad R_{net,sample-2,i}^{(2)}$$

In addition we let $\boldsymbol{\sigma(x_i)}$ and $\boldsymbol{\sigma(y_i)}$ denote the absolute standard deviations of $\mathbf{x_i}$ and $\mathbf{y_i}$ respectively. The two vectors of net count rates are denoted $\vec{x}$ **and** $\vec{y}$,

i.e. $\vec{x}$ **denotes the set of values** $\{x_i\}$ **and** $\vec{y}$ **denotes the values** $\{y_i\}$. The function measuring similarity between $\{x_i\}$ **and** $\{y_i\}$ is denoted by $Sim(\vec{x},\vec{y})$. The computation used for $Sim(\vec{x},\vec{y})$ is different depending on whether the same sample has been used or alternatively whether different samples have been used.

Case 1 : Same sample has been used for creating $\{x_i\}$ **and** $\{y_i\}$.

The computation of $Sim(\vec{x},\vec{y})$ is carried out as follows,

$$Sim(\vec{x},\vec{y}) = \sum_{i=1}^{K} \left\{ \frac{(x_i - y_i)^2}{\left[\sigma^2(x_i) + \sigma^2(y_i)\right]} \right\}$$

Case 2: Different samples have been used for creating $\{x_i\}$ *and* $\{y_i\}$**.**

The computation of $Sim(\vec{x},\vec{y})$ is carried out as follows,

Step 1. compute $\hat{\lambda} = \dfrac{\sum_{i=1}^{K} x_i}{\sum_{i=1}^{K} y_i}$

Step 2. then compute $Sim(\vec{x},\vec{y}) = \sum_{i=1}^{K} \left\{ \dfrac{(x_i - \hat{\lambda} y_i)^2}{\left[\sigma^2(x_i) + \hat{\lambda}^2 \sigma^2(y_i)\right]} \right\}$

Matching a Net Rate Vector from an unknown position

Here we consider that we have a net rate vector measured with the sample in any unknown position. This vector will be denoted $\vec{x}^u$. Locating an unknown position means taking $\vec{x}^u$ and searching in the archive of calibration data to find the "most similar" vector where most similar means having the smallest value of $Sim\left(\vec{x}^u, \vec{y}\right)$. The estimated location is the co-ordinates associated with the most similar vector.

In fact the archive search can be used to rank the calibration points in nearness to the unknown location. The co-ordinates of a set of 'near points' can then be used to provide a more refined estimation of sample position when the position is located between calibration points (and not exactly on one of them). In this paper however, we are presenting primarily data in which the unknown point is one of the calibration points.

4. Experimental Design and Data Management System

The underlying assumption of this approach is that in a situation without sample movement, there can be an exploitable relationship between the position of the sample and the net count rate vector coming from the detectors. In a practical application, the system of detectors would be designed to have an exploitable relationship. An effective design would address questions such as; how many detectors should be used, how efficient should they be and whether some of them should be distributed throughout the interior of the region of interest rather than all being attached to the perimeter wall. The answer to these questions would be aimed at ensuring discrimination between all profiles from sample locations that are not near to each other.

The principle of the method can however be tested in PERLA using the existing configuration of detectors even though this may not be optimal. To do this test, it is necessary to check the performance of the method using background and gross rate measurements made under controlled conditions. Apart from testing the feasibility this also gives some idea of the measurement time needed to provide a high probability of recognising the sample location. The necessary measurement times would of course vary with the set of detectors and their configuration. Checking that valid measurements are being used for a test of feasibility, means assuring that the stochastic process from each detector is a stationary process. This can be assured to some degree by ensuring absence of perturbing activities during the measurements. As well as this however, software tools have been used for reviewing time series from the detectors to screen them for any indication of non-stationary behaviour. This software uses a sequential detection module that has been developed for unattended measurement systems that need to recognise stationary background signals /6,7/. In the presence of a stationary process, the module uses rising and falling CUSUM tests /8/ to recognize any departure from the existing stationary behaviour. In the presence of a non-stationary signal, the module makes sequential use of non-parametric tests (e.g. Kendall rank test) to recognize a return to a stationary signal (perhaps with different mean). The module is adaptive in that the recognition of a stationary signal is accompanied by adaptive estimation of the mean and standard deviation to be employed in the two CUSUM tests. The detection module can be used to segment a time series into intervals of behaviour deemed stationary separated by intervals of behaviour deemed non-stationary. The module can be used in off-line mode to screen the data from the localization experiments in PERLA. It is applied separately to the time series provided by the signal of each detector. It verifies that the time series corresponding to a series of position measurements exhibit stationary signals in those time intervals in which the sample was immobile. The data of each stationary time interval provides a test measurement (i.e. $\vec{x}^u$) that can be used for testing the method.

A data management system was designed to take files of counts from a bank of detectors and then to analyse and use this data. Analysis of the data includes statistical and visual screening to ensure that during any measurement in a fixed position the detector time series were stationary. Use of the data includes creation of a calibration archive of profiles for different positions of measurement as well as using the calibration archive to locate the position of a sample. Location of a sample includes the use of the similarity index described earlier. This dedicated software was developed with the MS.Net

Developer Studio in Visual Basic. The system data are stored in an MS Access data file and the raw data are extracted from an ASCCI files.

The software includes functions:

- to measure and produce the calibration archive,
- to describe samples and the measurement geometry,
- to map information from ID reader for registering “ground truth” during experimental work,
- for real time localization or for batch processing of input files.
- to visualize the localization results.

The following figure shows a part of the user interface.

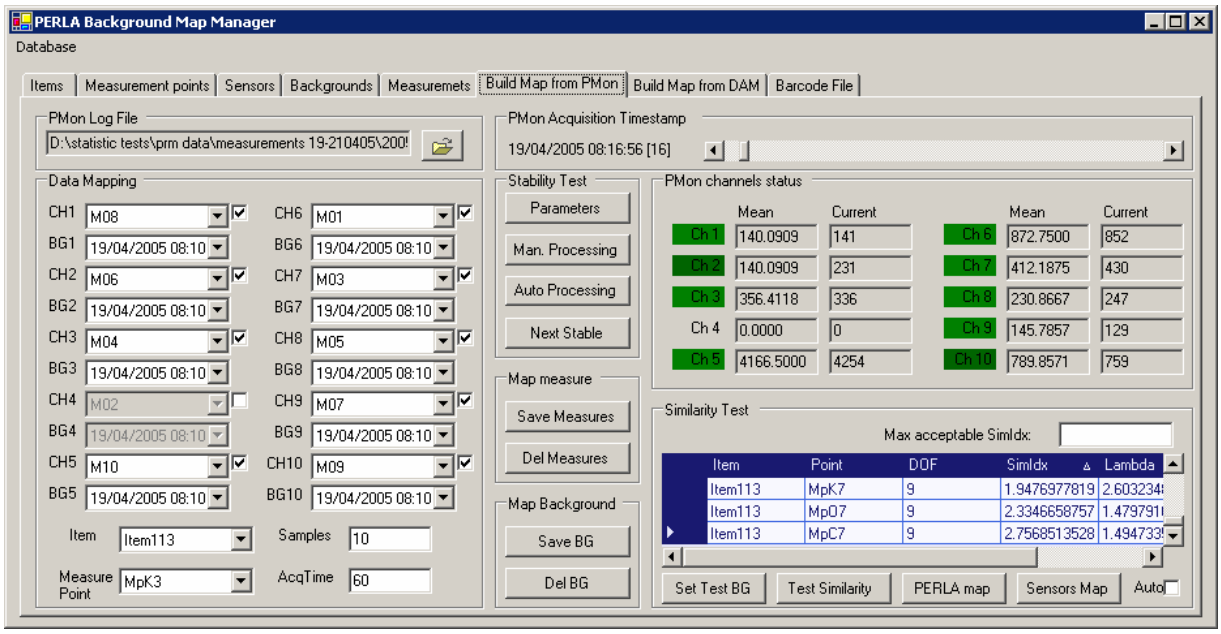

Figure 3: part of the user interface of the software prototype

The implementation of the calibration archive is described by the following parameters:

- For the NM item used to produce the calibration, an identifier, a name, a description and the weight,
- For the each calibration point in the archive an identifier, a name, a position x, a position y and a position z are stored, whereby the position z represents the height and is set to zero for the described experiments,
- For each detector an identifier, a name, a type, a comment, a position x, a position y and a position z,
- For the associated background measurements an identifier, the identifier of the sensor, count rate, the calibration identifier, the creation date, the counting time and the standard deviation,
- For the calibration archive measurements the identifier of the detector, the identifier of the measurement point, the identifier of the NM item, the count rate, a identifier of the calibration measurement, the creation date, the counting time, standard deviation, the identifier of the associated background measurement, the net count rate and the standard deviation of the net count rate,

These parameters are shown in the left hand-upper block of the user interface in Figure 3. The right hand side of the user interface shows the detector counts at a moment in time (most recent minute) and the similarity computations for the different hypothetical grid locations (for the 10 minute window). Figure 4 below shows the counts time series for each of the 10 sensors. The data stream for the ten minute window is outlined by means of yellow rectangles (hardly visible here). The green colour identifies a period in which the data behaviour is deemed stationary and red identifies an interval deemed non-stationary.

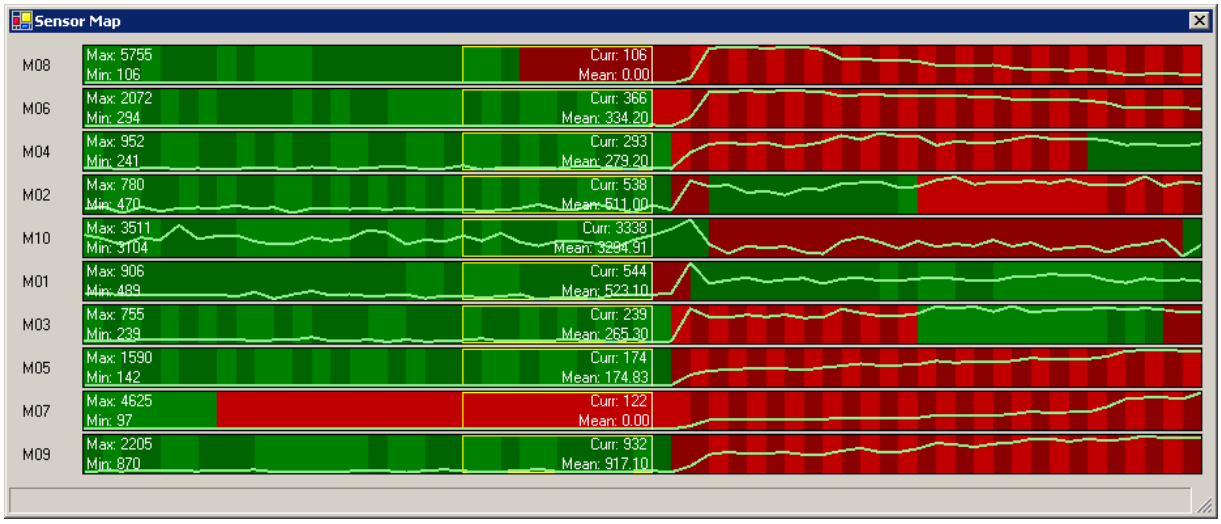

Figure 4: visualisation of the sensor raw data

In addition to the visualization shown in Figure 3 and Figure 4 the localization of a sample can be visualized in a 2D scheme of the PERLA laboratory by a point (as in Figure 5 below). Each detector point is coloured green or red according to whether the detector is currently deemed to be in a stationary period or not. The geographical spread of the colour blob gives a crude indication of the count rate at the detector. For this reason the detectors near the sample (M05) or those near the stores (e.g. M10 and M09) show higher count rates. The system can be informed that certain detectors are for some reason not in use. This type of information is used in the data processing and visualization (such as M02 under maintenance in Figure 5).

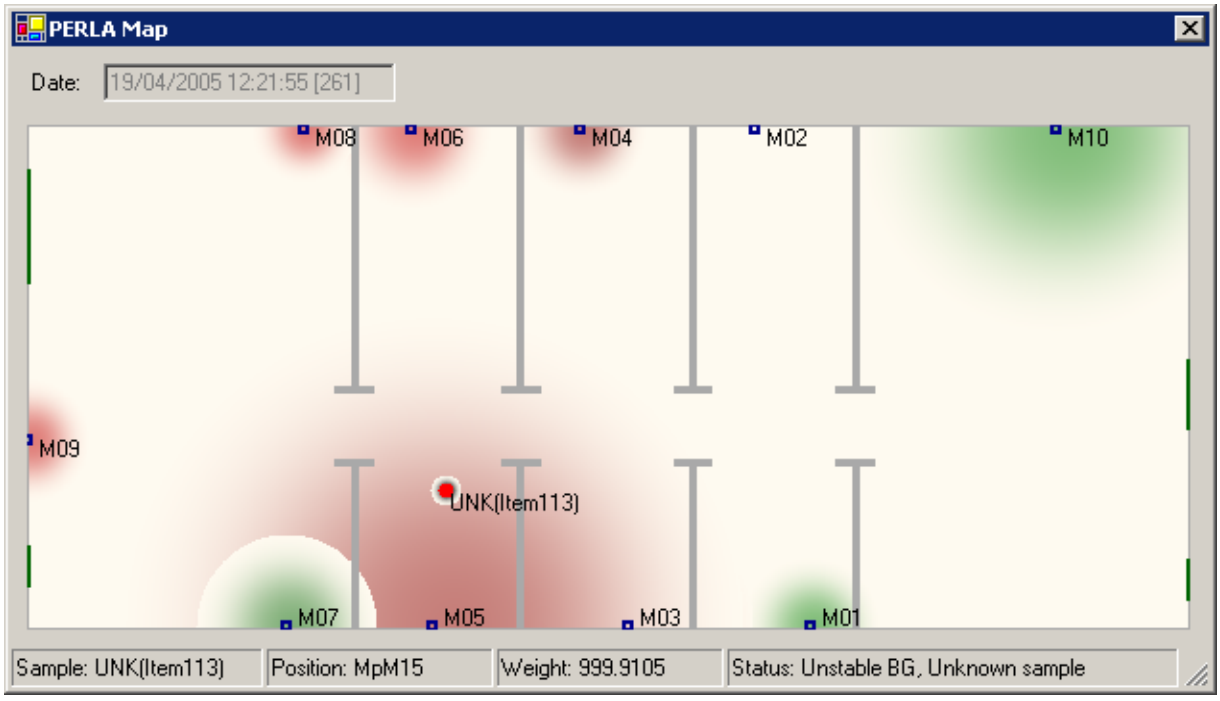

Figure 5: PERLA map with the result of the localization and the sensor status

5. Experimental Results

The experimental results are based on a calibration archive made in 2002 with measurement times of 3 minutes for the sample measurements (in the different positions) and 30 minutes for the background measurement. The results presented here refer to three series of test measurements (made in 2005)

using the calibration sample and two other samples. The localization experiments were performed on about 20 positions with 3 samples. All 3 samples are high burn-up PuO_2 samples. One sample (sample K) contained 400 grams of Pu, the sample (M) contained 1500 g of Pu and the 3rd was the calibration sample referred to earlier (L). Each sample is in double stainless steel containers.

Localisation of NM on points of the calibration archive

The software system has been used to record the detected neutron time-series at each detector during the whole duration of the test measurements - including the moments when a sample was being moved between two grid points. The primary counting interval for the detectors was 1 minute. The software system has then been used for off-line "real time" analysis of the continuous signal throughout the experiments. Moving through the data-stream the similarity index values for all the different positions were calculated on a fixed length moving time window (length 10 minutes). For each window, the system found the archive profile for which the similarity index is smallest and thus proposed an identified location for the sample at each moment. The time at which the sample was really placed in a particular location was noted and this was subtracted from the moment in time at which the similarity index analysis first identified the correct location. Figure 6 shows the elapsed time (in minutes) required for correct identification of the position for the 60 test measurements. The results show that the basic methodology works in the sense that the location of a sample can be recognised typically within 4 to 10 minutes. This detection performance is shown for three different samples of PuO_2 powder.

Of the 20 grid points used for localisation tests, 1 position was consistently not identified and all other were correctly identified. The unidentified position was due to an error in the calibration archive.

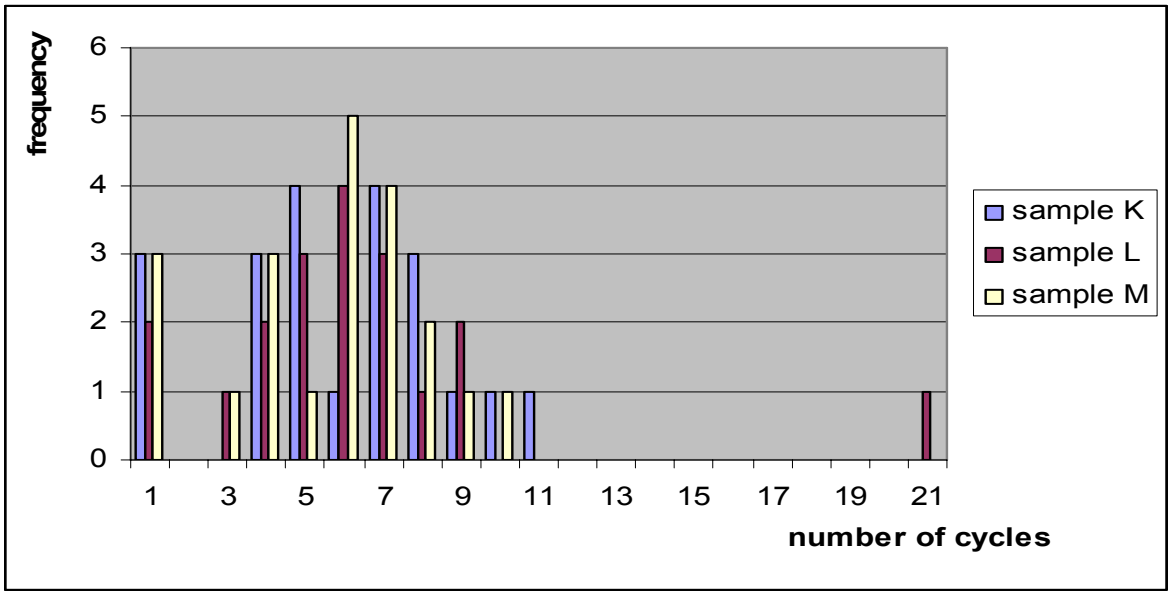

Figure 6: Histogram of elapsed time to correct localization for 60 measurements.

Localisation of NM on points not in the calibration archive

In addition to the measurements reported above, a small series of measurements (two different samples) were performed on 5 points which were not part of the calibration archive. The points were selected to be directly in between 2 calibration points. As shown in Table 1 below, the localization algorithm found the two calibration points closest to the true location. These were recognised by being the ones with the two smallest values of similarity index. The table also shows the elapsed time for this recognition process.

coordinate of the real position	Elapsed time (minutes) for sample K	Localized position best similarity index for K	Localized position second best similarity index for K	Elapsed time (minutes) for sample M	Localized position best similarity index for M	Localized position second best similarity index for M
2K	6	K3	K5	6	K3	K5
15J	8	I15	K15	15	I15	K15
15H	9	G15	I15	8	G15	I15
15F	9	E15	G15	8	E15	G15
15D	4	E15	C15	4	E15	C15

Table 1: localization of samples on points not part of the calibration archive

Tracking Movement

Visualisation of all the similarity index values through time showed that the minimum value increases as the sample moves away from its previous grid point and how the value for another point drops as the sample moves towards that point. This suggests that the approach can provide information for tracking movement – at least in direction. Whether this can provide a useful tracking capacity depends on the efficiency of the detection and the speed of movement.

6. Future Work

This paper has described an approach for the use of radiation patterns to identify the position of nuclear material. The approach is very robust because of the specificity of the calibration archive. The fact that background may vary spatially or that different regions may have very different neutron transport and absorption properties is represented in the archive profiles. Since the increased count rates due to the sample presence are compared with an archive of net count rates, the methodology requires knowledge of the background at the moment of placing the sample. This may be a difficulty if the background were changing significantly.

A number of questions need further study in assessing this approach for use in a monitoring system. These include,

- the number, type and locations of the detectors. The experiment was carried out in the PERLA laboratory using the existing neutron detectors in peripheral positions. A different pattern of location for the detectors (i.e. having some in the interior of the laboratory as well as some on the periphery) and using detectors with higher efficiency will reduce the time required to recognise of the position of the sample. Such increased sensitivity will also improve performance for detection of direction of movement. During this initial study, variation of these factors has not been studied.

- the experiment was aimed primarily at studying samples at locations which were one of the calibration grid points. Some results mentioned earlier show that when the sample is located somewhere in the interior of a square of grid points, the system can identify the near grid points. No study has yet been made of the use these “near grid points” to provide an estimate of the exact location of the sample.

- the effectiveness of the approach depends on the specificity of the calibration archive. Setting up this archive is a costly component of the method. If the laboratory environment changes as for example when large amounts of moderating material change position, the archive values may need to be adjusted to reflect the new situation. Adjusting the calibration archive to such events may require additional calibration measurements assisted by Monte-Carlo simulation of the new archive values.

Study of these questions will be the aim of further experiments in PERLA.

The authors would like to express their gratitude to their colleagues Dr Bent Pedersen and Dr Reinhard Berndt for useful discussion and advice when carrying out this experiment.

7. References

/1/ P. Lindén, J. K.-H. Karlsson, B. Dahl, I. Pázsit and G. Por; Localisation of a neutron source using measurements and calculations of the neutron flux and its gradient, Neutron Instruments and Methods in Physics Research A 438 (1999) 345-355.

/2/ S. Avdic, P. Lindén and I. Pázsit; Measurement of the neutron current and its use for the localization of a neutron source, Nuclear Instruments and Methods in Physics Research A 457 (2001) 607-616.

/3/ S. A. Pozzi, J.S. Neal and J.T. Mihalczo; Fission source location from passive nuclear materials identification system third order time correlation measurements, 45th INMM annual meeting, July 2004.

/4/ J. W. Howse, L. O. Ticknor, K. R. Muske, "Least squares estimation techniques for position tracking of radioactive sources" Automatica, November, 2001 Volume 37, Issue 11

/5/ A. Poucet, P. Schillebeeckx, et al. Recent improvements in PERLA and its evolution towards a European reference laboratory for NDA methods and techniques, Proc. of the 23rd ESARDA Symposium, Bruges, Belgium 8-10 May, 2001.

/6/ M. Boella et al.; Unattended measurement station (UMS): Towards the complete characterization of fresh LEU assemblies. Proc. 44th Annual Meeting of INMM, Phoenix, Arizona, USA, 13-17 July, 2003.

/7/ M. Franklin et al.; Unattended monitoring and measurement for safeguards, 15th Annual ESARDA Symposium, Rome, Italy, 11-13 May 1993.

/8/ D. M. Hawkins and D. H. Olwell; Cumulative sum charts and charting for quality improvement, Springer -Verlag New York, 1998.

Uncertainty Estimates in Spent Fuel Isotopics Using Post-Irradiation Examination of Spent Fuel Dissolution

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Abstract:

In an effort to strengthen safeguards and monitoring at reprocessing facilities, Texas A&M University and Los Alamos National Laboratory are collaborating on generating estimates of uncertainties in calculated spent fuel isotopics using sophisticated reactor physics simulations. These simulations use an integrated MCNP and ORIGEN code system for the accurate determination of spent fuel parameters (specifically spent fuel burnup and mass concentration of individual isotopes). The concentrations of a number of isotopes are calculated including uranium, higher actinide isotopes, and many fission products. The uncertainties associated with reactor models at different levels of fidelity are analyzed to determine the most simplistic approach possible while still maintaining a high degree of accuracy. These simulations are benchmarked to post-irradiation examinations of spent fuel dissolutions from PWR and BWR spent fuel to provide absolute uncertainty estimates. These results will aid in providing a cost-effective means of determining important aspects of spent nuclear fuel that would serve as an input to a nuclear material safeguards system at a reprocessing plant.

Keywords: nuclear material safeguards, spent fuel isotopic calculations, reprocessing

1. Introduction

The objective of this research was to facilitate the need for greater security at reprocessing facilities without a large commitment of financial resources. This research examined methods of using computer modelling to simulate the isotopic composition of spent fuel entering a facility and determine the amount of plutonium that might be present in that material. Relationships were also developed to determine the amount of plutonium based on the amount of ^{241}Am found in a sample of material. This can be a very important relationship as ^{241}Am can often be measured more easily than plutonium and with very little associated cost.

2. Background

The use of Monte Carlo codes have been widely accepted in applications such as flux calculations, but due to their large computational requirements have not been as accepted for

calculating isotopic concentrations. Most isotopic calculations have relied more on deterministic codes such as CASMO [1], HELIOS [2], SCALE [3], or ORIGEN [4].

MCNP (Monte Carlo N-Particle Transport) is a widespread Monte Carlo transport code used for stochastic simulation and the coupled transport of neutrons, photons, and electrons. MCNP can be used for a variety of applications including, but not limited to, dosimetry, radiation shielding, radiography, accelerator target design, and fission and fusion reactor design. The popularity of this code is largely due to its versatility, comprehensive geometry features, and its overall physics capabilities, including continuous energy treatment [5].

Monteburns (developed by LANL) [6] is a code that has been recently benchmarked [7, 8] for use in isotopic composition calculations. Monteburns is a Monte Carlo based neutronic code utilizing both MCNP and ORIGEN. MCNP serves as the transport solver and ORIGEN serves as the burnup module. Monteburns transfers one-group cross sections and flux values from MCNP to ORIGEN. The following equations demonstrate how the one-group fluxes and cross sections are generated with these codes. MCNP calculates one-group fluxes ($\bar{\phi}_i$) for any volume *i* by using the track length estimator of particle fluxes. One-group cross sections are calculated using track length estimators for reaction rates which essentially uses:

$$\bar{\sigma}_x^i = \frac{\int_0^\infty \int_{4\pi} N_i \sigma_x^i(E) \psi(\vec{\Omega}, E) d\Omega dE}{N_i \bar{\phi}} \quad (1)$$

Once the burnup and decay calculations have been performed by ORIGEN, Monteburns then transfers the isotopic compositions of the materials back to MCNP. Through the use of MCNP, Monteburns allows for the calculations of complex geometries and material compositions. This implies that Monteburns can simulate a vast array of different reactor types and is thus the code of choice for this project (where the type of material could be from nearly any type of reactor including thermal reactors, fast reactors, naval reactors, and research reactors).

3. Reactor Physics Model Development

This section describes the development of the Reactor Physics Model methodology. The first sections describe the reactor and measured data used to develop the models. The latter sections describe the various modelling parameters used to verify the statistical accuracy of the codes. This section concludes with a determination as to the required level of model fidelity for the nuclear forensics problem.

3.1 Takahama Unit #3 Test Case

The Takahama Unit #3 [9] reactor is a 3-loop pressurized water reactor (PWR) with an electric output of 870 MW and is operated by Kansai Electric Power Company (KEPCO). The reactor core contains 157 assemblies arranged in a cylindrical geometry. Each assembly contains a 17×17 square fuel matrix of which there are 264 fuel rods and 25 water holes. Of the 264 fuel rods, 14 of them contain 6.0 wt% gadolinium which is used as a burnable poison.

Samples were taken at various heights and measured by isotope dilution mass spectrometry. The isotope concentrations were then decay corrected to account for the cool-down time since being discharged from the reactor. Isotopes belonging to decay chains were corrected using Bateman's formula, while others were corrected using only their half-lives.

3.2 Modelling Parameters

Since MCNP is a Monte Carlo simulation code, the number of particles to be simulated was first determined. The accuracy of two parameters were considered: the critical eigenvalue (from Kcode calculation) and the scalar flux in the fuel (from an F4 track length estimator for the flux). A criticality simulation in MCNP consists of a specific number of particles per cycle, a total number of cycles, and a number of cycles to skip before recording results. The optimal level of each of these parameters was determined by iteration until the estimated uncertainty in the criticality and flux were both less than 0.1% with the smallest required computational time [10].

The Monteburns code utilizes three different input files: MCNP deck, Monteburns input deck, and irradiation history feed file. At this point, the MCNP input deck (containing geometry and material

composition) has already been created. The Monteburns input deck contains information on the individual isotopes to be tallied, power (MW) of the model, and the number of the burnup steps in the feed file. Most of this information is taken directly from the reactor data given in the literature [9].

The isotopes to be tallied in Monteburns consisted of the isotopes of interest and a standard set of actinides. Tallying these additional actinides improves the overall accuracy of the code by allowing Monteburns to update the one-group cross section sets for various reactions. It should be noted that while tallying all of the isotopes for which there are libraries would significantly increase the accuracy of the code, the tremendous increase in computational time would far outweigh the benefits. For this reason, extra isotope tallies must be chosen very carefully.

The feed file contains the irradiation history of the fuel. The feed file allows the user to specify as few or many burnup steps as the situation requires. While more burnup steps (resulting in lower burnup per step) are desirable for purposes of accuracy, each step requires additional computational time. A convergence test on the size of the burnup step was performed in previous research [10] and found that 2,500 MWd/MT was the optimal value.

3.3 Best Estimate Model

The results of the Reactor Physics Model methodology showed conclusively that the advanced 2D pin cell provided the greatest level of accuracy while maintaining a minimum degree of computational time. Several correction factors were also found to be needed in order to improve the accuracy of the model. It is well known that neutron cross sections vary with changes in temperature. Some of these effects include doppler broadening of the cross section resonances, change in density of the moderator, and the thermal neutron scattering effects. The thermal neutron scattering effects are included through the use of an S(α,β) treatment in MCNP. This generates neutron cross sections (particularly for nuclides such as hydrogen) for neutron energies less than 4 eV.

As axial locations are not known in these models, the average moderator temperature in the core was used to determine the density.

The S(α,β) tally consisted of an “mt” card in the MCNP input file. The “mt” card chosen was the lwtr.62t. This cross section was created from the ENDF/B-VI Rev 3 and the sab2002 library and is for use specifically with hydrogen in light water at a temperature of 600K.

For the fuel temperature correction factor, a review of the available cross sections in the MCNP library found a cross section that better fit the environment of the model. The cross sections chosen were 92235.15c, 92238.15c, and 94239.15c. These three cross sections were created from the endf62mt library with a temperature of 800K.

Another correction factor used was the addition of U^{234} and U^{236} to the initial fuel isotopics. While this information may not be known for the Forensics problem, there are a set of equations that can be used to accurately predict the concentrations of these isotopes based on the enrichment of the fuel. These equations are as follows:

$$^{234}U\ wt\% = 0.0089 \times {}^{235}U\ wt\% \quad (2)$$

$$^{236}U\ wt\% = 0.0046 \times {}^{235}U\ wt\% \quad (3)$$

where ^{X}U is an isotopic designation and *wt%* is the weight percent of that isotope with respect to the rest of the fuel. While ^{236}U does not exist in nature, it should be noted that the inclusion of ^{236}U isotopes in fresh fuel is only for U.S. born fresh fuel. This occurs because U.S. enrichment plants are contaminated with ^{236}U due to a previous processing of naval reactor spent fuel through the plants. Table 1 shows the RMS percent uncertainty for the isotopes of interest using the best estimate model.

	Best Estimate Model
^{235}U	1.56
^{238}U	0.04
^{238}Pu	8.25
^{239}Pu	4.41
^{240}Pu	2.58
^{241}Pu	6.81
^{237}Np	6.14
^{137}Cs	1.65
^{148}Nd	0.64
^{154}Eu	16.42
^{241}Am	30.25
^{242}Am	50.18
^{243}Am	10.04

Table 1: RMS Percent Error of Isotopes of Interest.

4. Reactor Physics Model Benchmarking

This section describes the Reactor Physics Model benchmarking process for the methodology developed in the previous section. The first part describes the various reactors and their respective data used to benchmark the methodology. The last sections describe the results of the benchmarking.

4.1 Calvert Cliffs Unit #1

Calvert Cliffs is a Combustion Engineering designed 2-loop PWR operating at 883 MW electric. There are 390 assemblies in a rectangular array, each containing a 14×14 square lattice of fuel rods and waterholes. The enrichment of the fuel varies from 2.05 wt% to 2.99 wt% depending on the location in the core [11]. Isotopic dilution mass spectrometry was used to examine the samples. The isotopic concentrations reported from this reactor were not decay corrected to the time of discharge. This was very beneficial as complex production and decay chains can add unwanted uncertainties to the final results.

4.2 Trino Vercelles Unit #2

Trino Vercelles is a Westinghouse designed PWR operating at 825 MW electric. The core has fuel enrichment from 2.719 wt% to 3.897 wt% depending of the location in the core. The fuel assemblies contain a 15×15 square lattice of 208 fuel rods. The remaining space in the assembly is taken by cruciform control blades [12, 13]. The analyses of this reactor were performed at two separate facilities, Karlsruhe Laboratory and Ispra Laboratory, and their results compared. Each of the labs performed radiochemical analyses on the samples that included both alpha and gamma spectrometry. The reported isotopic concentrations were decay corrected from the time of discharge from the reactor. It should also be noted that there was a three year shutdown in the middle of the fuel cycle for the fuel being examined. This shutdown time had a considerable effect on the ability to accurately predict isotopes such as ^{241}Am.

4.3 Benchmarking Results

The benchmarking of the Reactor Physics Model methodology demonstrated good agreement of the actinide isotopic concentrations being evaluated. The percent error in isotopic concentration for each reactor as well as the total of these values is shown in Table 2. As can be seen, the total error associated with each actinide was less than 5% with the exception of ^{237}Np. It should be noted that the measured uncertainty of this isotope was reported as +/-10%. As the results are within that measured uncertainty, this model was probably predicting ^{237}Np better than the numbers show. Further studies will have to be done to verify this. ^{154}Eu should be removed from the list of isotopes of interest as it is a very difficult isotope to accurately measure with its complex production chain. Two isotopes, ^{238}U and ^{148}Nd, had total errors of less than 1%, which is exceptional. This is very important as ^{148}Nd is the primary isotope for determining the burnup of the fuel. If the burnup of the fuel is wrong, that error will propagate throughout the entire system. The three isotopes of americium that were examined showed varying results. Further inspection shows that large uncertainties showed in the Takahama and Trino Vercelles reactors are likely due to errors in the decay correction process. The results from the Calvert Cliffs reactor were, however, very promising. The reported data from this reactor was not decay corrected and showed an uncertainty for ^{241}Am of only 1.41%.

	Takahama	Calvert Cliffs	Trino	Total
^{235}U	1.56	3.55	0.7	1.31
^{238}U	0.04	0.02	1.48	0.49
^{237}Np	6.14			6.14
^{238}Pu	8.25	2.68		4.34
^{239}Pu	4.41	1.67	3.15	1.89
^{240}Pu	2.58	3.92	0.85	1.59
^{241}Pu	6.81	1.21	3.84	2.64
^{137}Cs	1.65			1.65
^{148}Nd	0.64	1.75	0.34	0.63
^{154}Eu	16.42		6.99	8.92
^{241}Am	30.25	1.41	67.99	24.81
^{242}Am	50.18		173.35	90.23
^{243}Am	10.04	7.7	13.69	6.21

Table 2: RMS Percent Error for Benchmarked Isotopics.

5. Discussion of Results

^{241}Am and ^{243}Am were further inspected to determine their abilities to predict plutonium. Both of these isotopes were compared to total plutonium in the system as well as plutonium quality. It was found that ^{241}Am was much more accurate in predicting plutonium concentrations than ^{243}Am. It was found that the amount of ^{243}Am, when compared to plutonium, would rise sharply at first and then saturate and level off. Both the sharp rise and level saturation make it exceedingly difficult to determine plutonium concentrations with a reasonable level of accuracy. The concentration of ^{241}Am, compared to plutonium, rose at a more steady rate than that of ^{243}Am. Figure 1 shows ^{241}Am compared to plutonium quality and figure 2 shows ^{241}Am compared to total plutonium concentration. As can be seen from figures 1 and 2, ^{241}Am showed a more consistent relationship (almost a steady 45 degree increase) with the quality of plutonium than with the total plutonium in the system.

The relationship of ^{241}Am to plutonium can be utilized in several different facets of the reprocessing process. The first use would obviously be in the cutting of the fuel rods. A gas mass spectrometer could be used to measure ^{241}Am and thus determine the amount of plutonium at that stage of the process. The other scenario would be if a detector system had malfunctioned and it was not discovered until the fuel had already left the facility. In the event that this would occur, an analysis could be conducted on the uranium waste stream (or waste tank) to determine the amount of ^{241}Am and thus the amount of plutonium that was present in the fuel.

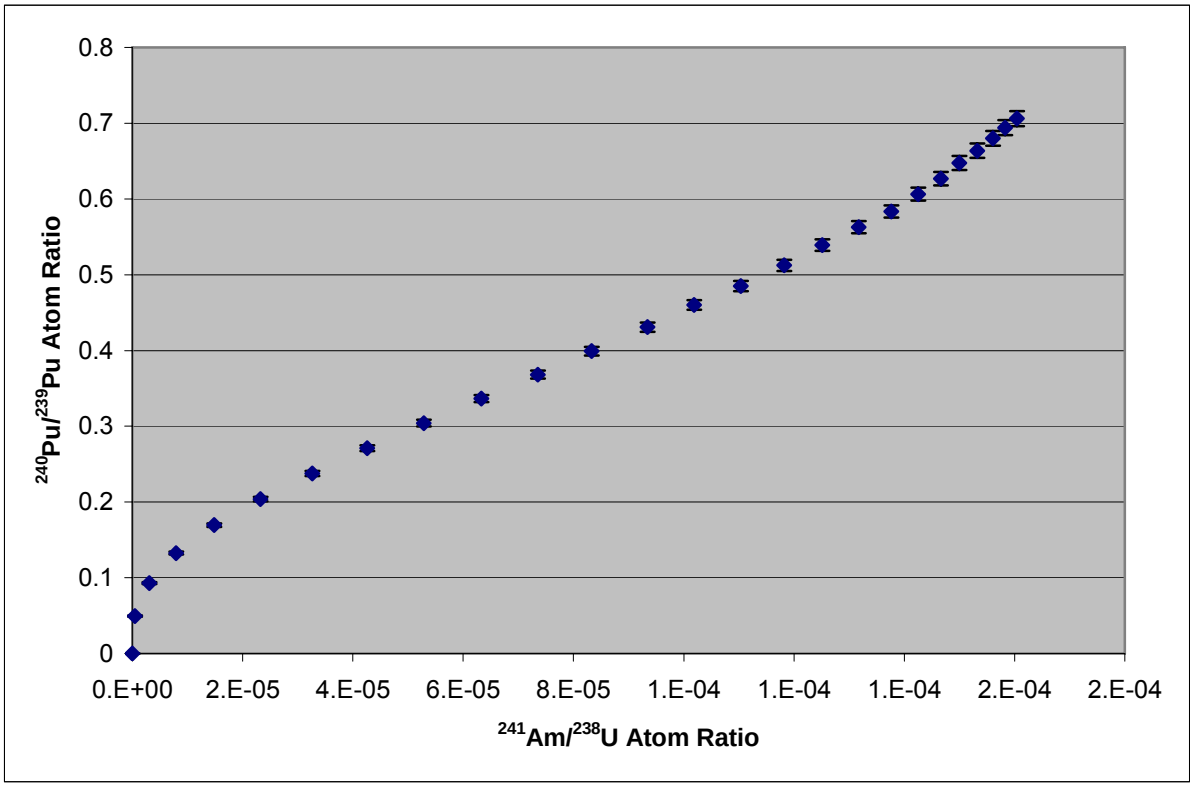

Figure 1: Plutonium Quality Versus ^{241}Am For Calvert Cliffs Unit #1.

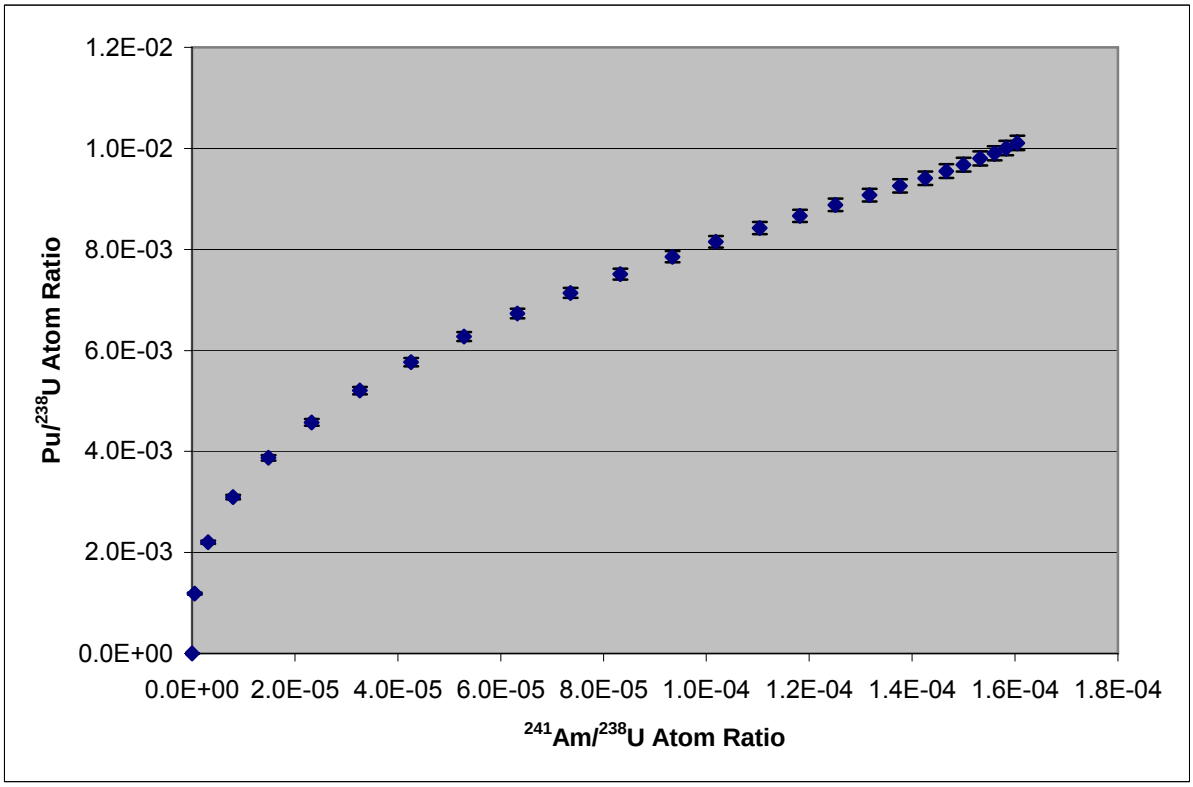

Figure 2: Total Plutonium Versus ^{241}Am For Calvert Cliffs Unit #1

6. Conclusions

The results from the Reactor Physics Model methodology showed that the advanced 2D pin cell gave the greatest degree of accuracy while minimizing computational requirements. The correction factors employed were as follows: appropriate fuel temperature cross section file, moderator density, S(α,β) tally, ^{234}U and ^{236}U initial fuel concentration, and ^{153}Eu tally if the ^{154}Eu isotope is being examined. The results demonstrated that future research into this area should choose reported data that has not been decay corrected. The decay correction techniques do not accurately represent isotopes with complex production and decay chains. While both ^{241}Am and ^{243}Am showed reasonable levels of uncertainty for the Calvert Cliffs reactor, ^{241}Am is more accurate at predicting the concentration of plutonium (especially the quality of the plutonium).

The measuring of gases such as ^{241}Am is already in practice at several reprocessing facilities around the world. This process is extremely inexpensive when compared to the costs of mass spectrometry on actinides such as plutonium. Proper utilization of the methodology described here could help minimize costs while increasing safeguards through the use of computer modelling and gas sampling. One stage in particular that this methodology could be useful is in the cutting of the fuel rods. A verification of the amount of plutonium present at this stage could help determine if materials were diverted before dissolved. Another possible use for this methodology includes the determination of the amount of plutonium in fuel that has already left the facility. This determination could be made by analyzing the amount of ^{241}Am in the uranium waste stream (or waste tank).

7. References

[1] M. Edenius, K. Ekberg, and B.H. Forssén, "CASMO-4 - A Fuel Assembly Burnup Program - User's Manual," Studsvik Report SOA-95/1, Studsvik of America (1995).

[2] E.A. Villarino, R.J.J. Stamm'ler, and A.A. Ferri, "HELIOS: Angularly Dependent Collision Probabilities," *Nuclear Science and Engineering*, **112**, 16-31 (1992).

[3] "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation," NUREG/CR-0200, Rev. 6 (ORNL/NUREG/CSD-2/R6), Vols. I, II, and III, Nuclear Regulatory Commission (2000).

[4] A.G. Croff, "A User's Manual for the ORIGEN2 Computer Code," ORNL/TM-7175, Oak Ridge National Laboratory (1980).

[5] J.F. Briesmeister, "MCNP – A General Monte Carlo N-Particle Transport Code Version 4B," LA-12625-M, Los Alamos National Laboratory (1997).

[6] D.L. Poston, and H.R. Trellue, "User's Manual, Version 2.0 for MONTEBURNS Version 1.0," LA-UR-99-4999, Los Alamos National Laboratory (1999).

[7] W.S. Charlton, R.T. Perry, B.L. Fearey, and T.A. Parish, "Calculated Actinide and Fission Product Concentration Ratios for Gaseous Effluent Monitoring Using Monteburns 3.01," *Nuclear Technology*, **131**, 1-18 (2000).

[8] W.S. Charlton, W.D. Stanbro, and R.T. Perry, "Comparisons of HELIOS, ORIGEN2, and Monteburns Calculated ^{241}Am and ^{243}Am Concentrations to Measured Values for PWR, BWR, and VVER Spent Fuel," *J. Nuclear Science and Technology*, **37(7)**, 615-623 (2000).

[9] Y. Nakahara, "Nuclide Composition Benchmark Data Set for Verifying Burnup Codes on Spent Light Water Reactor Fuels," *Nuclear Technology*, **137**, 111-125 (2002).

[10] D.E. Burk, "Forward Model Calculations for Determining Isotopic Compositions of Materials Used in a Radiological Dispersal Device," Master's Thesis, Texas A&M University, 2005.

[11] S.R. Pati and P.A. VanSaun, “Isotopics and Transuranic Nuclide Content of Three- and Four-Cycle Calvert Cliffs-1 Fuel,” Research Project 1755-1, Combustion Engineering (1982).

[12] “Isotopic and Criticality Validation for PWR Actinide-Only Burnup Credit,” DOE/RW-0497, U.S. Department of Energy (1997).

[13] S. Guardini and G. Guzzi, “Benchmark Reference Data on Post Irraditaion Analysis of Light Water Reactor Fuel Samples,” EUR-7879-EN, Nuclear Science and Technology, Commission of the European Communities (1983).

Forensics Technique for Attributing Unknown Spent Nuclear Fuel

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Abstract:

If a radiological dispersal device (RDD) is detonated in the U.S. or near U.S. interests overseas, it will be crucial that the actors involved in the event can be identified quickly. In the case where spent nuclear fuel is used as material for the RDD, law enforcement officials will need information on the origin of the spent fuel. This information will then be used to help identify the specific individuals who manufactured the device and perpetrated the event. Texas A&M University and Los Alamos National Laboratory are collaborating on the development of a technique for identifying the material used in a radiological dispersal device. The methodology developed makes use of both a forward model and an inverse model to identify specific spent fuel characteristics using isotopic composition of RDD debris. The forward model is based on sophisticated reactor physics calculations for the prediction of spent fuel isotopic compositions as a function of fuel type (e.g., PWR, BWR, CANDU, RBMK, etc.), fuel burnup (in MWd/MTHM), fuel age (in years since permanent discharge from the reactor), and operating characteristics (e.g., total shutdown time). These reactor physics calculations are benchmarked to measured data to establish their accuracy in predicting isotopic compositions. The inverse model makes use of measurements of actinide and fission product isotopic ratios in the RDD debris. A description of the inverse model will be given, accuracies of the technique, and the results to date.

Keywords: spent fuel; nuclear; forensics; attribution; terrorism

1. Introduction

Terrorism has demanded that the U.S. be prepared for many situations that it never considered a significant threat before. One of these threats is the possibility of an organization detonating an RDD on U.S. soil. The ability to quickly identify the source of the radiological material used in an RDD would aid investigators in identifying the perpetrators. If spent fuel is used as the radiological material for the weapon, it could prove to be deadly and cause extensive disruption. In this work, we developed and implemented a forensics methodology to attribute spent fuel to a source reactor assembly. The specific attributes determined are the spent fuel burnup, age from discharge, reactor type, and initial fuel enrichment. Determining the attributes of spent fuel is only part of a larger effort [1]. The forensic project as a whole includes the process from time of measurement until verification of the suspected spent fuel. This paper is only one part of this entire process. Once the attributes of the spent fuel are determined by the inverse process, these attributes are compared to the world's reactors. If a match is found a forward model is created and used to verify the match.

2. Monitor Development

Each fuel characteristic of interest to the forensics problem uses an isotopic monitor. The fuel characteristics of interest here are fuel burnup, fuel enrichment, reactor type, and fuel age. Each characteristic has unique requirements that the monitor must meet. A specific set of monitors must be

used for determination of the enrichment. Burnup, reactor type and age determination can be performed using any combination of isotopes that meet the requirements.

2.1. Burnup

The nuclear industry has been interested in fuel burnup for decades and has developed accurate methods to measure burnup [2]. Burnup is the amount of energy produced per kg of fuel. At nuclear power plants, initially the fissions from ^{235}U and ^{238}U provide the source of neutrons and power in the reactor. As the reactor burns the fuel, ^{236}U, ^{239}Pu, ^{240}Pu and several other minor fissionable actinides are created and fissioned which then contribute to the neutron flux. For economic reasons, nuclear power plants want to increase the fuel burnup as high as possible. After the fuel is discharged from the reactor, the burnup can be measured indirectly by measuring a burnup monitor. Coupled with mass spectrometry, it has been shown that burnup can be measured within one percent accuracy [3].

A burnup monitor can be any fission product that is created directly proportional to the burnup. For the forensics problem a more stringent restriction exists because the reactor type is unknown. Therefore, we need burnup monitors that are produced at the same rate regardless of the reactor type. The neutron energy spectrum and fissioning isotopes are primarily what changes from one reactor type to another. So for our purpose a burnup monitor is any nuclide whose fission yield is constant as a function of the neutron flux energy and fissioning isotopes. By identifying a burnup monitor that has a constant yield across reactor types, the burnup may be determined without knowing additional information about the fuel. To avoid the effects of decay, burnup monitors that have half-lives longer than 200 years should be chosen. Another concern is transmutation of the burnup monitor, a burnup monitor needs to have a small radiative capture cross section to avoid the destruction of the burnup monitor as it is produced. Having a large radiative capture cross section introduces reactor-dependent terms in the burnup calculation. To find the burnup without knowing other reactor parameters, it is necessary to avoid these terms.

The production of an isotope in a reactor is usually dependent upon more than its direct fission yield. Isotopes are also produced by the decay of other fission products. An isotope's decay chain can have a large impact on its production if its cumulative yield is significantly higher than its direct yield. This requires all of the fission products in a burnup monitor's decay chain to exhibit the same properties as the burnup monitor. The exception to this would be in the case where the fission products individual yield is so small that it doesn't significantly effect the production of the burnup monitor. All of the isotopes in the decay chain must have a half-life shorter than a few days to avoid delayed production of the burnup monitor after the spent fuel is discharged. There are a few isotopes that are shielded by a stable isotope and thus have no decay chain. If a shielded isotope meets the requirements of a burnup monitor, then the complications of a decay chain will not apply.

To find burnup, an iteration scheme with the following two equations is used:

$$BU(T) = \left[\frac{N^{B}(T)}{N^{U238}(T)}\right]\left[\frac{N^{U238}(T)}{N_o^{U}}\right]\frac{N_a E_R}{Y_B} \tag{1}$$

and

$$\left[\frac{N_o^{U}}{N^{U238}(T)}\right] = \frac{\left[\frac{N^{U}(T)}{N^{U238}(T)}\right] + \left[\frac{N^{Pu239}(T)}{N^{U238}(T)}\right] + \ldots}{1 - \frac{M_o^{U}}{N_a E_R} BU(T)} \tag{2}$$

Where T is the time at discharge, *BU* is the fuel burnup, N^B is the atom density of isotope *B*, M_o^U is the initial uranium mass, E_R is the average energy per fission, N_a is Avogadro's number. The numerator on the right hand side includes all fissioning isotopes.

First we use Eq. 2 and set *BU(T)* to zero, then iterate between Eq. 1 and Eq. 2. The burnup equation is not very sensitive to changes in the initial uranium atom density, allowing it to converge after only a few iterations.

2.1. Enrichment

Determining the initial enrichment after the spent fuel has been discharged is a new problem. Fuel is very well characterized before being used in a nuclear power plant and there has been no need in the past to determine the initial enrichment from spent fuel. If no other fissionable isotopes are present while the fuel is in the reactor, then the enrichment would change linearly with burnup. In reality, several other fissionable isotopes are produced and burned, thus complicating the determination of the initial enrichment.

There are several reasonable assumptions made when calculating the enrichment. We assume that:

1. The only isotopes that fission are ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu and ^{241}Pu.
2. There is no production of ^{235}U and ^{238}U.
3. All one group cross sections are constant with time.
4. ^{239}Np and ^{239}U decay instantaneously to ^{239}Pu.
5. The decay of all fissionable isotopes are neglected.

The only isotope that has a short enough half-life to significantly effect our calculation is ^{241}Pu. ^{241}Pu has a 14.1 year half-life, which will cause our calculation for initial enrichment to be slightly high. The higher the burnup and the longer the fuel has been discharged, the larger the effect will be on our results.

The initial enrichment is first predicted by using an equation developed from balance equation of the uranium atom density. The equation is over a page long but can be found at the following reference [1]. Benchmarking showed that the initial enrichment predicted had errors as high as 15% which is to high for accurate attribution. An addition forward model iteration step was implemented to improve the enrichment prediction. An ORIGEN model of each reactor type being considered is built and ran. The ^{235}U content from the model is compared with the measured value. The initial enrichment and burnup is then adjusted and the model is ran again. This process is repeated as shown in Fig 1 until the ^{235}U content matches the measured ^{235}U. By adding this extra step the initial enrichment error is reduced to less than 2%.

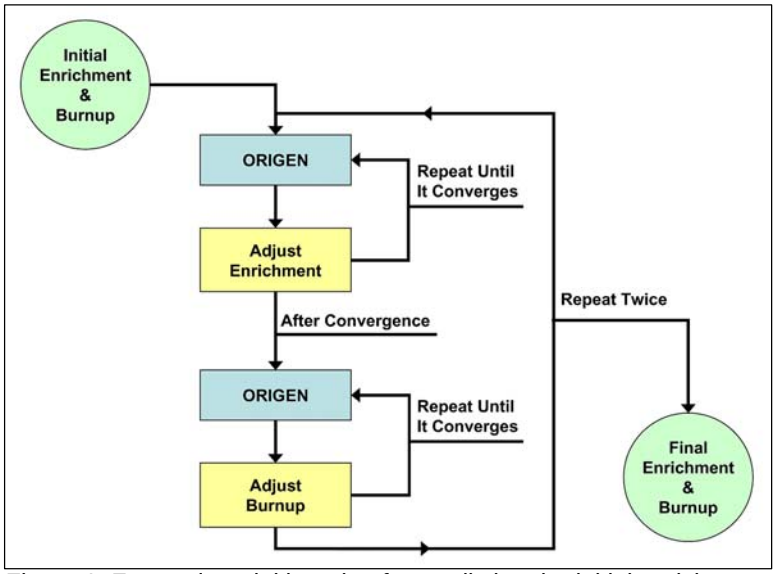

Figure 1. Forward model iteration for predicting the initial enrichment.

2.2. Reactor Type

Determining the reactor type is critical to the success of this project. Our ability to predict the reactor type relies heavily on the accuracy of the one-group cross sections and yield values. The cross sections are created from MCNP reactor core models. To differentiate the reactor types from each other, isotopes with cross sections and yields that change significantly from reactor type to reactor type are needed. To avoid the complication of decay, stable or long-lived isotopes are needed. The method for determining reactor type depends upon the accuracy of ORIGEN, and as the decay chain becomes more complicated the accuracy is diminished. ORIGEN also has a more difficult time tallying isotopes in trace amounts with a high degree of accuracy. To avoid these issues, it is best if the chosen isotopes have a simple decay chain and are produced in quantities above 0.01 moles per metric ton of fuel.

The mass spectroscopy measurements are generally performed as isotopic ratios. ^{238}U is the most abundant isotope by many orders of magnitude in spent fuel. By measuring a reactor type monitor with respect to ^{238}U, slight changes of the quantity produced of the monitor will make little difference in the measured value. To avoid this and improve the accuracy, reactor type monitors are measured with respect to something that is produced in similar quantities. This will ensure that slight changes in reactor type monitor quantities will be easily evaluated. This is especial true when the burnup is low.

2.3. Age

Once the burnup, initial enrichment and reactor type are known, finding the time since the discharge date is relatively straightforward. The method requires ORIGEN and the accuracy of the solution depends upon the accuracy of ORIGEN. Age monitors should be chosen by their half-life and the ability of ORIGEN to accurately calculate them. To get the best accuracy, an isotope with a half-life close to the age of the spent fuel is used to make the final calculation. Age monitors that have a half-life between one and thirty years should be chosen, since the oldest spent fuel is less than 60 years old.

At this point, five reactor types that match the reactor type monitors the best are chosen. For these five reactor types an ORIGEN model is built for each. ORIGEN runs the models and only allows the spent fuel to decay for approximately 30 days. This allows the short-lived isotopes to completely decay into the more stable isotopes. The isotopic quantity is collected for each of the age monitors from the ORIGEN output. This value is then set as the initial quantity, N_o^C, which is the quantity of isotope C at the time of discharge. Equation 3 is a rearrangement of the standard decay equation in measurable terms.

$$T = -\frac{T_{1/2}}{\ln(2)} \ln\left(\left[\frac{N^C(T)}{N^{U238}} \right] \Big/ \left[\frac{N_o^C}{N^{U238}} \right] \right) \tag{3}$$

The age is then calculated for each age monitor and averaged. Not all of the age calculations from each age monitor are going to be accurate. The closer the half-life of the age monitor is to the actual age the more accurate it will be. The average age of all the age monitors should be close to the actual age. Once an initial average is calculated the two age monitors that have the closest half-life to the initial calculated average age are selected. Then a new average is calculated with just these two age monitors. This is then reported as the age, or the time passed since the spent fuel was discharged from the reactor. The age determination relies upon the correct calculation of burnup, initial enrichment, and reactor type. Therefore, if there are large errors in any of these calculations, it will affect the age prediction.

If ^{241}Pu is used as an age monitor, problems may arise with mixed oxide (MOX) fuels. This may cause the initial amount of ^{241}Pu, or to be more than the present-day measured amount which will cause the age to be negative. In this event the ^{241}Pu should be removed as an age monitor and not used. Another factor that will affect the age prediction is the shutdown time. Fuel is usually used in cycles and in between cycles the fuel is temporarily stored onsite. During the shutdown the age monitors will decay and cause the ORIGEN forward model to incorrectly calculate the atom density of the age monitor at the time of discharge.

2.3. Suggested Monitors

A summary table listing all of the suggested monitor nuclides for the nuclear forensics problem is given in Table I.

Table I.
Suggested Monitor Nuclides

Burnup Monitors	^{140}Ce, ^{100}Mo, ^{98}Mo, ^{97}Mo, ^{138}Ba, ^{142}Ce, ^{148}Nd
Enrichment Monitors	^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu
Reactor Type Monitors	^{109}Ag, ^{153}Eu, ^{156}Gd, ^{143}Nd, ^{240}Pu, ^{108}Cd, ^{113}Cd, ^{149}Sm, ^{166}Er, ^{132}Ba, ^{98}Tc, ^{115}In, ^{72}Ge, ^{115}Sn
Age Monitors	^{90}Sr, ^{93}Nb, ^{106}Ru, ^{101}Rh, ^{102}Rh, ^{125}Sb, ^{134}Cs, ^{137}Cs, ^{146}Pm, ^{147}Pm

3. NEMASYS

A Microsoft Visual Studio .NET 2003, Visual Basic code was written to implement the methodology described. This code was named the Nuclear Event Material Attribution SYStem, or NEMASYS. NEMASYS is GUI (graphical user interface) based and user-friendly. The code was written on a Microsoft Windows XP platform and requires Microsoft's .NET framework for proper execution.

The code needed to meet several goals and requirements. The first goal was to provide a way for a user who only has a basic understanding of nuclear engineering principles the ability to obtain useful information in identifying the radiological source in an RDD. The code needed to be easy to operate. The code also needed the ability to provide all of the necessary information a technical person would want. This would give someone who is familiar with the process and theory behind the code the ability to check the conclusions and make adjustments as necessary.

NEMASYS is only needed in times of emergencies and, therefore, will not be used on a regular basis. Thus if an event occurs NEMASYS must be useable regardless of whether or not trained personnel are available. Therefore, it needs to run in an intuitive manner that is user friendly. There will not be time for the user to read a large manual or spend days understanding how the process works. If an event occurs, the user will have a few days at best to process the data and produce a detailed report on their findings.

Before NEMASYS can perform any calculation, it must first load all the necessary information about each reactor and the list of isotopes to be used as monitors. The list of monitors and reactor types are pulled from a Microsoft Access database. This was done to give the user the ability to add new monitors and reactor types without having to do any re-coding. A text input file is also used to store the location of the files needed for each reactor type. This includes the cross-section and yield data files and a skeleton ORIGEN model. It is stored in a text file to allow SENTRY the ability to modify and change the location of the data and the data itself. For example, if a better cross-section set is found, a user could replace the old cross-section set through a SENTRY interface (Lotus Notes). A screen shot of a simple NEMASYS report can be seen in Fig 2.

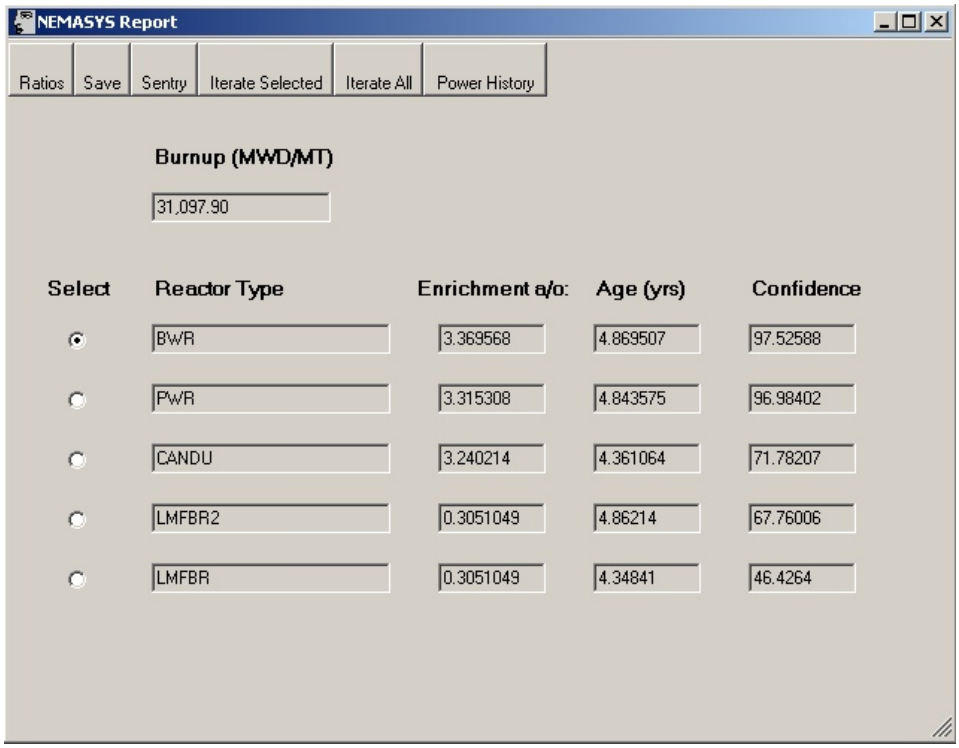

Figure 2. Initial reactor report from NEMASYS.

4. BENCHMARKING NEMASYS

Benchmarking the results is critical to validating the methodology and process used. Unfortunately, there is limited data available on the isotopic composition of spent fuel. Obtaining spent fuel composition measurements requires enormous resources. The data that is available is limited mainly to actinide content and a few fission products. A generally complete listing of possible isotopic spent fuel measurements can be found in the NEA data bank [4]. A complete set of data that met the minimal requirements for analysis by NEMASYS was found for a reactor in Japan. The measurements were from the Mihama Unit 3 reactor.

The Mihama nuclear reactor is located in the Fukui prefecture in Japan. Nine samples were taken from three different spent fuel assemblies from the third unit (Mihama-3). Three samples were taken from the first assembly (JPNNM3SFA1), two samples from the second assembly (JPNNM3SFA2), and four samples from the third assembly (JPNNM3SFA3). For the reactor type, there was no fission product data available to differentiate a PWR from a BWR. The accuracy of the reactor type prediction is thus limited due to this lack of available data. The power history of each assembly is shown in Table II.

The burnup was found to be within 4.5% or less of the reported burnup. The error associated with the reported burnup was not given with the data. Without knowing the exact method and technique of the burnup measurement the error is unknown. Most likely the error was at least 5% showing that our methodology is within their error. The enrichment calculation was within 1.25% or less which also suggests that our burnup prediction is very reasonable. The age was reported to be 5 years, which is most likely an approximation and not exact. The age prediction by NEMASYS had errors that ranged from 1% to 10%. The age is heavily dependent upon the previous predictions and the ORGEN models, and therefore is the most susceptible to error propagation. NEMASYS easily predicted the correct reactor type in every case if you lump BWR and PWR in the same group. The BWRs and PWRs were lumped together because a reactor type monitor that could distinguish between PWR and BWR was not given in the available data. Therefore the ability to distinguish between BWR and PWR was not verified with actual data.

5. CONCLUSION

This work is only a small part in a larger effort to attribute spent fuel. Additional work on NEMASYS and on the complete attribution project still needs to be completed. Additional benchmark data would allow for improvements in the results and building additional confidence in NEMASYS. There are still opportunities to improve the accuracy of the methodology as well. This can be done by automating the process to find outlier monitors, and performing an error analysis of the processes used to find the burnup, enrichment, reactor type and age. Once a more stringent analysis of the methodology is complete, the calculation of error propagation needs to be implemented in NEMASYS. With sufficient benchmarking results, each monitor can be studied in greater detail and verified to work as predicted. This can only be done after more spent fuel measurements are performed. The implementation of the shutdown monitor has yet to be incorporated into NEMASYS. Not only will knowing the shutdown time help with matching the suspected fuel to a power history, but it might aid in the calculation of the age.

Only a few reactor types were considered in our study. This needs to be expanded to include all commercial and research reactors. Large commercial sources are not considered by NEMASYS, and a complete study needs to be performed to allow the implementation of a forensics methodology for large commercial sources.

It has been shown that post-event material attribution is possible with enough accuracy to be useful for investigators. The burnup can be found to within 5% accuracy, enrichment to within 2% accuracy, and age to within 10% accuracy. Tabel II compares the values reported with the measured data and the predicted values. The methodology can discriminate between a CANDU, LMFBR, BWR and PWR. It was implemented into a code call NEMASYS. NEMASYS was written so that its capabilities could be expanded and improved without the need to perform any re-coding. New reactor types and monitors can be added and defined by the user. This gives the user the ability to fine tune the process as more data becomes available. NEMASYS is easy to use, and it takes a minimum amount of time to learn its basic functions. It will process data within a few minutes and provide detailed information about the results and conclusions. In conclusion, NEMASYS has enough accuracy to provide useful results but would still benefit from evolutionary improvements by more data for benchmarking and modifications to the methodology.

Table II.
Comparison of the Results From NEMASYS to the Reported Data

Sample	Burnup (MWd/MT)		Initial Enrichment (a/o)		Age (Years)	
	Reported	Predicted	Reported	Predicted	Reported	Predicted
1	8,300	7,952	3.25	3.22	5	4.85
2	6,900	6,678	3.25	3.27	5	4.81
3	15,300	14,664	3.24	3.20	5	4.70
4	21,200	20,399	3.24	3.27	5	4.78
5	14,600	14,043	3.24	3.20	5	4.52
6	29,400	28,394	3.25	3.21	5	4.97
7	32,300	30,931	3.25	3.29	5	4.85
8	33,700	32,371	3.25	3.22	5	5.07
9	34,100	32,920	3.25	3.21	5	5.01

6. References

[1] M. Scott, *Nuclear Forensics: Attributing the Source of Spent Fuel Used in an RDD Event*, M.S. Thesis, Texas A&M University, 2005.

[2] A.V. Bushuev, A.F. Kozhin, G. Li, V.N. Zubarev, A.A. Portnov, V. P. Alferov, and M.V. Shchurovskaya, *Determination of the Fuel-assembly Burnup in a Research Reactor by Repeated Short-time Irradiation Followed by G Spectrometric Measurements*, Atomic Energy,97, 2, 2004.

[3] J.J. Giglio, D.G. Cummings, M.M. Michlik, P.S. Goodall, and S.G. Johnson, *Determination of Burnup in Spent Nuclear Fuel by Application of Fiber Optic High-resolution Inductively Coupled Plasma Atomic Emission Spectroscopy*, Nuclear Instruments and Methods in Physics Research A 396, 251, 1997.

[4] "List of Reactors in SFCOMPO on WWW Ver.2," http://www.nea.fr/html/science/wpncs/sfcompo/Ver.2/Eng, Nuclear Energy Agency, 2005.

A tool set for the quick and efficient exploration of large document collections

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Abstract:

We are presenting a set of multilingual text analysis tools that can help analysts in any field to explore large document collections quickly in order to determine whether the documents contain information of interest, and to find the relevant text passages. The automatic tool, which currently exists as a fully functional prototype, is expected to be particularly useful when users repeatedly have to sieve through large collections of documents such as those downloaded automatically from the internet. The proposed system takes a whole document collection as input, carries out some automatic analysis tasks, annotates the texts with the generated meta-information, stores the meta-information in a database, and provides the users with an interface that allows them to search and view the most pertinent text passages. In the first step, named entities (names of people, organisations and places) are recognised and stored. Then, highly similar documents are grouped into clusters of documents that are likely to report about the same event or subject domain. In the third step, domain-related terms are detected and extracted from each cluster of documents. Then, a zoomable map is generated for each of the document clusters to show the geographic coverage of the collected documents. Additionally, names and specialist terminology found in the document cluster are listed. Hyperlinks allow the users to jump to the text passage where they were found, and also to read the individual documents that are part of the cluster. All clusters of the document collection are made available in an html file that can be viewed with a standard web browser. When the system is used on a regular basis, it builds up a historical database that contains information on which names have been mentioned together with which other names or places, and users can query this database to retrieve information extracted in the past.

Keywords: text analysis; multilinguality; clustering; named entity recognition; automatic knowledge acquisition; visualisation; information overflow.

1. Introduction

Most information-seekers and analysts feel that the so-called *information overflow* is a major restriction to their work. Although a lot of useful information is freely available on the internet and from other sources, it gets increasingly difficult to *find* the relevant information. Search engines are getting better at identifying the most relevant documents on the internet, but users still need to look through many documents to find the piece of information they need.

We are proposing a solution that helps information seekers to sieve through large collections of documents quickly and to speed up the process of finding what they are looking for. Assuming that users already have a large document collection to look through, the solution consists of automatic text analysis software that firstly clusters all related documents into groups, and that then extracts information from each cluster and presents it to the users in an organised and intuitive manner. For each cluster, the extracted information consists of: (a) a list of keywords that indicate the approximate contents of the document cluster, (b) a list of geographical places mentioned in the cluster, lists of (c) names of people and (d) of organisations, and (e) a list of the user interest-specific spe-

cialist terms found in the cluster. In principle, further information such as lists of all dates, addresses, telephone numbers, etc. could be extracted and displayed, as well. Figure 1 shows how extracted information for a given cluster of documents can be presented to the user.

The proposed system currently exists as a prototype that is specialised in *news* analysis, i.e. on the processing of thousands of news articles every day. Part of the system can be tested at the address http://press.jrc.it/NewsExplorer.

The application consists of many individual components which cannot all be described here in detail. The purpose of this article is to show the potential of such a system, without going into too many technical details and without always comparing the individual applications with the state-of-the-art. Instead, we will refer to some of our more specialised publications where the technology used is described in more detail.

Following this introduction, we first describe ways of gathering potentially user-relevant documents and of preparing them for an automatic analysis, including the identification of document duplicates and near-duplicates (section 2). We then dedicate some space to the analysis tasks of clustering related articles (sect. 3) and of extracting keywords, places, names and terms from them (sect. 4). Section 5 and 6 are dedicated to the visualisation and presentation of the extracted text data, as well as to ways of helping users cope with foreign language text collections. Section 7 gives information on the usability status of the prototype software.

2. Gathering and pre-processing the document collection

Information-seeking users may either already have collections of documents that they need to look through, or they may use automatic document gathering tools that automatically collect documents that are potentially relevant. Commercial web crawlers that download documents from specific web sites exist, but it is also possible to develop a tool that exploits the power of existing web search engines like Google by downloading all the documents found by the search engine. Furthermore, there are crawlers that specialise in specific document types such as news. An example of such a system is the *Europe Media Monitor* EMM (http://emm.jrc.org; Best et al. 2002), which monitors about 20000 news in thirty languages a day.

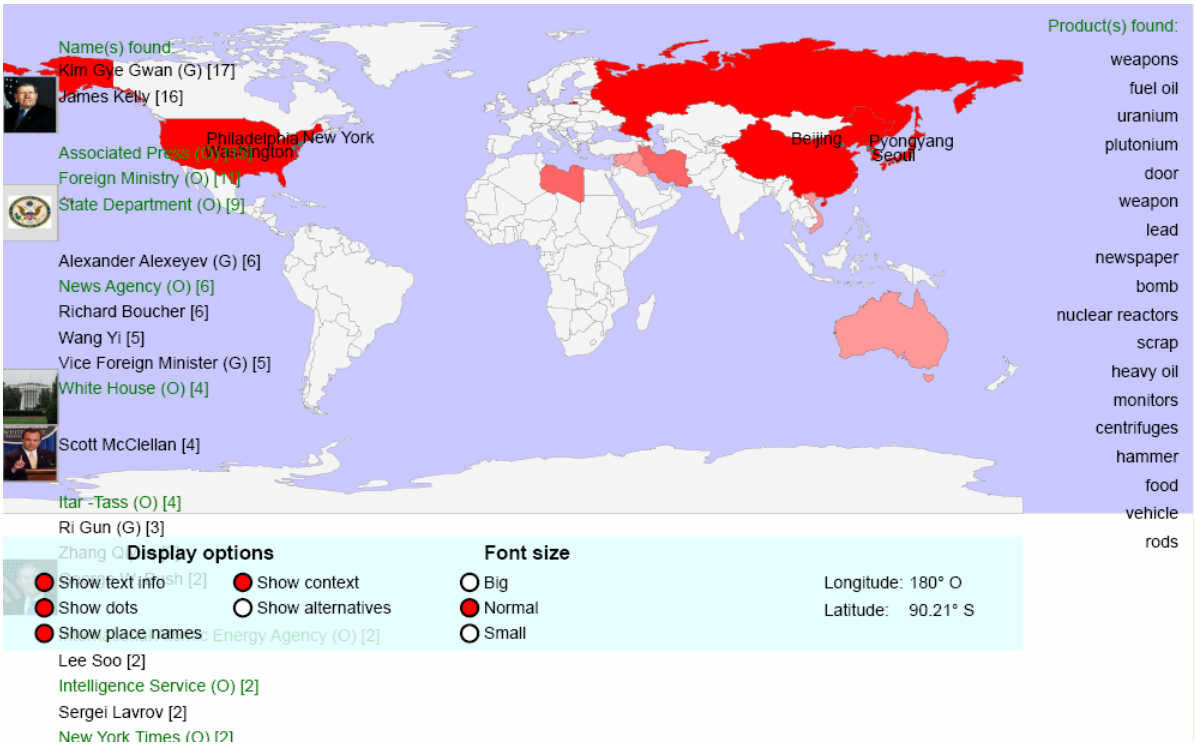

Figure 1. Automatically generated map showing the geographical coverage of one news cluster on North Korea, consisting of 26 articles from 10 different newspapers. The map also shows specialist terms as well as the names of people and organisations automatically extracted from the text collection. Hyperlinks on the places and on the terms allow users to jump directly to the relevant text passages, while a click on any of the names leads to a database containing additional information on the people and organisation.

2.1. Document format conversion

In which ever way the collection is being compiled, it is likely that the documents are in different file formats, such as HTML, PDF, MS-Word, postscript, plain text, etc. Automatic text processing tools require that the text be available in machine-readable format such as plain text or, ideally, in a more structured format such as XML. Tools that take proprietary file formats and convert them to plain text or XML are commercially available. For news analysis, the standardised XML subtype RSS is appropriate, as it is widely used (Hammersley 2003).

2.2. Identification of document duplicates and near-duplicates

There is a certain likelihood that the document collection contains several copies of the same or very similar documents. At the *Joint Research Centre*, we are using a duplicate identification software tool that has been adapted from software developed at the German *Institut für Deutsche Sprache* in Mannheim. The tool splits every document of the collection into word pentagrams (i.e. five consecutive words) and checks, for each pair of documents, whether some of the pentagrams are identical. If more than a certain percentage of all pentagrams are identical (e.g. 50% or more), this can be flagged to the user or the duplicate documents can automatically be removed, depending on the users' preferences. The tool thus identifies whether entire texts or some text passages are *identical* between two documents. By doing this, multiple reading of the same text can be avoided to save time or to avoid duplicate counting of the phenomena the information seeker is looking for.

3. Grouping of documents according to similarity

Several non-identical documents are likely to describe the same contents. In news analysis, for instance, we found that a main event (such as the Madrid Bombing in March 2004 or the death of Pope Jean-Paul II in April 2005) can cause the publication of over one hundred news articles per day in English alone. Some of them are partial duplicates (e.g. all news articles based on the same news wire), others simply talk about the same things. It is useful for the information seeker to know which documents are in such a way related in order to either discard them altogether (if they are not of interest) or to concentrate their effort on them.

The grouping of related documents can be automated in a statistical process called *clustering*. In the clustering process, software calculates the similarity between each pair of documents. All those document pairs or document groups that are more similar than a certain threshold can then be grouped into a single cluster (see Pouliquen et al. 2004b). The following sections explain how we represent documents by their keywords and the countries mentioned in them, and how we then calculate the similarity between the documents.

3.1. Keyword identification for document representation

To produce a meaningful and machine-readable representation of each document, we represent each text by a ranked list of its most pertinent words (the *keywords*) and their relative importance (the *keyness* of the keywords). The approach to produce keyword lists for a new document exploits the fact that words have an average frequency in texts. For instance, the word 'the' occurs about 60000 times per million words, while the words 'table' and 'xenophobia' occur about 200 and 1 times per million, respectively. If a word occurs considerably more frequently in a given text than it occurs on average (normalised by the total number of words), then this word is likely to be of some importance, i.e. it is a keyword. *Keywords* are thus those words that occur outstandingly frequently in the given text. *Keyness* is the degree with which they are outstandingly frequent. Both keywords and their keyness can be calculated

Keyword	Keyness
cluster	714
name	361
jrc	334
keyword	320
tool	317
pouliquen	290
place_name	264
figure	264
2004	238
language	224
keyness	187
2003	187
ignat	169
2005	162
collection	160
multilingual	155
analysis	145
steinberger	141
table	137
display	128
word	125
automatically	124
news	113
…	…

Table 1. Sample document representation of the document you are currently reading, i.e. a list of keywords and their keyness.

with statistical formulae such as TF.IDF or the *log-likelihood* or *chi-square* tests (See Kilgarriff 1996). At the JRC, we use the *log*-likelihood formula to calculate which words are keywords, as it seems to perform best for our kind of data (Kilgarriff 1996). Table 1 shows the top of the automatically produced list of keywords for the document you are currently reading.

3.2. Particular consideration of geographical information

In order to distinguish news about similar events (such as election campaigns or reports on earth quakes) in different countries, we add geographical information to the list of keywords. We do this by first identifying references to places in each article (see section 4.2), and by adding to the country score of the article each time we find a reference to a city or town of that country. As some countries generally are being talked about much more than others, we furthermore normalise the country score by comparing the count with average counts, using again the log-likelihood formula. This means that if a document mentions 'France' and the cities of 'Paris', 'Lyon' and 'Tours', this is counted as having four references to the country 'France', and this number is normalised by the average number of references to 'France' in texts of a similar type (e.g. news). The list of a document's keywords is thus enhanced by the country ISO code 'FR' and its keyness value. The resulting list of keywords and country codes, together with their keyness values, can be seen as a vector that represents the document.

3.3. Document similarity calculation and clustering

Similarity calculation can be based on lexical overlap or other measures. Lexical overlap means that two documents are judged to be similar if they are made up of the same or similar words. This means that the more words they have in common, the more they are judged to be similar. High-frequency words that occur in all sorts of documents such as *the, of, for, from*, etc. (*stop words*) are usually not considered in the comparison. At the JRC, we base our similarity calculation on the vectors produced in the steps described in sections 3.1 and 3.2.

The similarity between two documents can then be calculated as the *cosine* between the vector representations of the two documents. The cosine expresses the similarity or distance of two vectors: If the cosine is 1, the vectors (and thus the documents) are identical. If the cosine is close to zero, the vectors are very dissimilar.

The grouping of related documents can be automated in a statistical process. In this clustering process, software calculates the similarity between each pair of documents. Documents that are more similar than a certain threshold can then be grouped into one cluster. For the analysis of collections of news articles, we found empirically that the threshold of 0.5 produces meaningful results (Pouliquen et al., 2004b).

For each cluster, the most typical document is identified, i.e. the document that is most similar to the centroid in the document cluster's vector space. The title of this centroid article is used as the name for the cluster.

4. Extracting information from each cluster of documents

In our current application, each cluster of related texts is henceforth treated as if it were a single (meta-) document about a certain event or story. For this meta-document, we automatically extract and present information. We currently identify keywords (section 4.1), information about geographical references (4.2), and information about references to people and organisations (4.3). Where available, we also search for the occurrence of terms from a user-controlled term list in the document (4.4). We plan to extract and display further information in the future, including dates, products, etc.

4.1. Keywords

For each cluster, we identify and display the main keywords in order to give users an overview of the contents of the cluster. In order to calculate the cluster's keywords, we take as input the keywords of each individual document, which were calculated in the process described in section 3.1. We then calculate the average keyness of each keyword across all individual articles to come up with the averaged keyness (*group averaging*). The keywords with the highest averaged keyness are then displayed to the users.

4.2. Geographical references

A geo-coding tool automatically recognises references to countries, cities, towns and villages. Rules and heuristics are used to distinguish place names from homographic person names and normal words (there are places called 'And' 'Split', 'Bush', 'Victoria', etc.), and to decide which of several homographic place names is being talked about in a text (world-wide, there are 45 places called Paris, 18 places called 'Birmingham', etc.). The result is a list of geographical place names found in the (meta)-document, together with information on the cor-

responding country and with exact latitude and longitude information. A separate map generation tool produces zoomable interactive SVG maps (Eisenberg 2002) in which the countries and cities mentioned are shown (see Figures 1 and 2). Colour intensity is used to indicate the number of hits per country. When users point the mouse on a highlighted place or country, one-line of left and right hand side contexts of the place name are shown (similarly to Figure 3 for terms). The idea of this application was to let users investigate quickly in which context the place name was mentioned. The details of this application are described in Pouliquen et al. (2004a) and Ignat et al. (2003).

4.3. Person and organisation names

A name recognition tool also scans each (meta)-document to identify references to people and organisations. The current version of the software was tuned for news analysis. It identifies uppercase words in text as names either if they are already known names stored in a database, of if the context makes it clear that they are names. These contextual patterns are expressions that can be titles (Professor, President, Lithuanian Minister for Foreign Affairs), professions (painter, artist) or other trigger words ('has said',). For English alone, the list contains about 1300 manually compiled and empirically derived context patterns. In one year of news analysis, we have found about 125000 distinct names. Every day, an average of 500 new names is added to the database.

In the current version of the JRC tool for the exploration of document collections, all names found in a given cluster are listed at the left hand side of the map showing the geographical coverage of the same document cluster (see Figure 1). Additionally, the trigger words that helped to identify the name can be displayed next to the name (e.g. 'US Secretary of State' or 'Iranian religious leader'). This is deemed useful to provide additional information on the person identified.

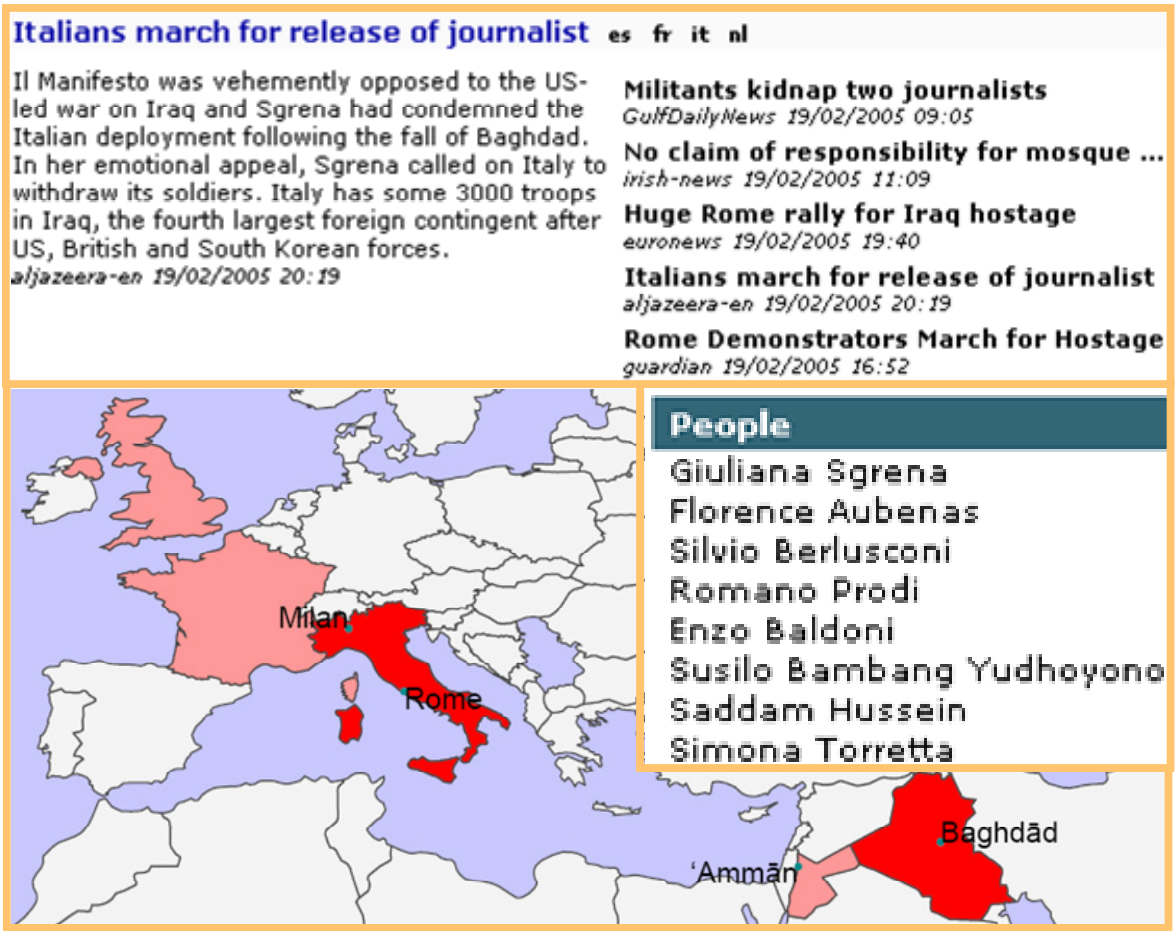

Figure 2. English news cluster with automatically identified links to related news clusters in Spanish, French, Italian and Dutch. Hyperlinks on the titles allow users to read the individual articles, links on names allow them to find out more about the persons, all news articles in which they are mentioned, etc.

m programme. Since then it has also restarted its ***plutonium*** programme. "We have a saying: 'Fool me once,
to a plutonium one. It has offered to freeze the ***plutonium*** programme, but the US says a freeze is not enough
s an enriched uranium programme, in addition to a ***plutonium*** one. It has offered to freeze the plutonium prog
m programme. Since then it has also restarted its ***plutonium*** programme. "We have a saying: 'Fool me once,
to a plutonium one. It has offered to freeze the ***plutonium*** programme, but the US says a freeze is not enough
s an enriched uranium programme, in addition to a ***plutonium*** one. It has offered to freeze the plutonium prog
Korea is believed to have supplied weapons-grade ***plutonium*** to Pakistan and to have obtained uranium-bas

Figure 3. All occurrences of the term 'plutonium' found in a cluster of news articles, and their context.

4.3.1. Identifying name variants

As the documents in a cluster normally come from different sources, the same name is often written in different ways (see Table 2). This is particularly true for Arabic and Russian names, as there are various transcription standards. For some names, we found over sixty variants in one year of news analysis. As it is useful to combine the variants under one single alias, or even a numerical person identifier, we automatically launch an approximate matching tool for all names found in a cluster. This tool splits each name into all possible sets of consecutive two-letter and three-letter groups (bigrams and trigrams, more generally *n-grams*) and then calculates the similarity between the n-grams of two names, using the cosine formula. When the overlap between two names is above a certain threshold, i.e. when two names have a large percentage of n-grams in common, then they are flagged as being variants of the same name. In order to be able to handle languages with different alphabets (Arabic and Russian), additional algorithms have been developed. For details on the name recognition and the approximate name matching work, see Pouliquen et al. (2005). The automatically identified variants are stored in a database to build up a list of all variants for the same name that have been found over time. Furthermore, all articles and clusters mentioning any of the variants are indexed with the numerical name identifier so that users can search the database for all articles about a certain person, independently of how the name was spelled. As the approximate matching sometimes misses variants, or it maps two names that are not the same, the automatically achieved results can be checked interactively and they can be corrected.

Mohammed ElBaradei
Mohamed El Baradei
Muhammad al-Baradai
Mohammed al-Baradei
Mohamed al-Baradei
Mohammed El Baradei
Mohamed El-Baradei
Mohammed el-Baradei
Mohamed el-Baradei
Mohamed el Baradei
Mohamed ElBaradei
Mohammed el Baradei

http://de.wikipedia.org/wiki/Mohammed_el-Baradei
http://en.wikipedia.org/wiki/Mohamed_ElBaradei
http://fr.wikipedia.org/wiki/Mohamed_ElBaradei
http://it.wikipedia.org/wiki/Mohamed_ElBaradei

Table 2. Name variants found in many different news articles and automatically identified as belonging to the person *Mohammed ElBaradei*. List of Wikipedia encyclopaedia entries automatically found and displayed to give users quick access to information about a person.

4.3.2. Identifying related persons

The JRC system keeps track of all person and organisation names mentioned in the same clusters. When names are mentioned repeatedly over time, a network of related persons and organisations thus develops, because some names will co-occur more often in the same clusters than others. The system learns over time, as it is updated daily. For a given person, users can thus ask the system to display lists of those persons that are most frequently mentioned in the same clusters. A more sophisticated weighting technique also allows to display mostly those names that are specifically associated to a given person, i.e. those that mainly occur with this person and not so frequently with other persons. For details, see Pouliquen et al. (2005).

4.3.3. Finding information on the internet

As information on some people is available in online encyclopaedias on the internet (e.g. http://en.wikipedia.org), we automatically check for each name identified whether information about this person is available. If it is, we store the URL of this page so that we can make it available to the user (see Table 2). That way, users can find additional information about a person with a simple click.

nuclear [39]
weapons of mass destruction [12]
uranium [7] (*natural uranium|uranium*)
plutonium [7] (*plutonium*)
disarmament [6]
missiles [5]
iaea [4]
missile [4]
proliferation [3]
atomic [2]
biological [1]
reprocessing [1]
enrichment [1]
nuke [1]
scud [1]

Table 3. List of specialist terms found in a cluster of six news articles, and their frequency.

4.4. Specialist terms

Depending on their job-specific interests, users may be looking for different terms in texts, such as nuclear terminology, names of diseases, financial terms, etc. If they can see, for a given cluster of tens of documents, all of their terms of interest that occur anywhere in the cluster, they can quickly decide whether the document cluster is of interest, what aspect of their work is covered by the group of documents, etc.

The current system therefore allows users to provide term lists with tens, hundreds or even thousands of terms of their field. The JRC tool then searches each cluster automatically for occurrences of any of the terms and lists them for the user (see Table 3). When clicking on a term in the term lists, a separate window opens to show the complete list of contexts (see Figure 3). Users can thus see the context of each of the terms without having to open the file in which the term occurred. It is planned to add a further hyperlink allowing the users to jump directly to a certain document in case the small context looks interesting, to allow a further exploration of the document.

Semi-automatic creation of domain-specific term lists

For the automatic display of the user-relevant terms found in a given cluster, long lists of potentially relevant terms are required. Users do not normally have such lists available, and potentially existing domain glossaries may not be suitable for the task as they often only contain the standardised version of terms, while other versions (synonyms or alternative spellings) may exist and may be used in real-life texts.

In order to quickly compile large lists of domain-specific terms as they occur in real-life texts, we use tools that automatically extract all potential terms from specialist documents provided by the users. These terms can then be ranked by domain specificity, using statistical methods. Users can then simply look through the resulting term lists and choose those words or word groups that they would like to be highlighted in texts. It is up to the users to decide whether they are interested in having only highly specific terms identified in a document cluster, or whether they also want to know about more general terms. The term lists in Figure 1 and Table 3 also contain rather general terms such as ‘rods’ and ‘monitors’.

The terms can additionally be marked as belonging to different sub-areas, and this additional information can be displayed together with the terms. In the field of *Nuclear Safeguards*, for instance, terms could be divided according to the part of the nuclear fuel cycle they relate to. Enrichment-related terminology, for instance, could then be identified even more quickly.

It is our mid-term aim to provide users with term lists from various fields of interest (e.g. medicine, finance, computer science, etc.), from which they can choose. Once this aim is achieved, they can simply tick subject area boxes and the terms from the chosen fields will be displayed.

4.5. Language-specific issues

It is relatively easy to detect the occurrence of a term or of a place name in text if the word form found in the text is the same as the word form found in the list of terms or place names. However, special means need to be used, for instance, to find the plural form of a term in text (e.g. *centrifuges*) if the term list contains the singular term (*centrifuge*). For the English language, the problem is reduced to plural and to genitive (*centrifuge’s*) forms. In more highly inflected languages such as those of the Slavic or Finno-Ugric language families, the number of word variants can be rather large. In Slovene, for example, the noun *centrifuga* can appear in any of the following forms: *centrifuga, centrifugi, centrifugo, centrifug, centrifugama, centrifugah, centrifugam, centrifugami.* In Russian, the following word forms for the equivalence of centrifuge can be found: центрифуга, центрифуги, центрифуге, центрифугой, центрифугу, центрифуги, центрифугам, центрифуг, центрифугами, центрифугах.

This problem can be solved by using morphological processing tools such as lemmatisers, which analyse the inflected word form and re-

turn the uninflected or neutral dictionary form of the term. However, such linguistic tools are not available for all languages, and purchasing them for a large number of languages is rather expensive. At the JRC, we therefore tackle the problem by using simple heuristics and exhaustive lists of suffixes and their combination with the word stems. In Russian, for example, the suffix list for the stem 'центрифуг-' (*centrifuge)* contains the suffixes: 'а|и|е|е|ой|у|ам|ами|ах'. These lists are used to generate all term variations so that the term will be successfully found in text. For details, see Steinberger et al. (2004).

5. Organisation and visualisation

The results of the analysis of the whole document collection is a list of document clusters, where additional information is available for each cluster (see Figure 2). The JRC's system orders the clusters by size (largest first) and displays all the extracted information: cluster title, map showing the geographical coverage of the documents in this cluster, and the lists of names, organisations and terms. Alternatively, related clusters can be displayed together, independently of their size. The users can thus start exploring the document collection by clusters. They can decide whether to sieve through the collection by size, or whether they first want to identify those clusters that seem most promising because they contain many relevant terms or names, etc.

When the users click on any of the names, they will see a page showing the name, the name variants found, the list of encyclopaedia entries found (if any, see Table 2), a list of all articles and of all clusters mentioning this person, and a list of related names and organisations (see section 4.3.2). Users can thus explore and navigate the document collection, for instance by querying the database for all news clusters in which a person has been mentioned. They can find clusters mentioning a certain keyword or country, etc., and they can ask for further related clusters.

6. Overcoming the language barrier

A bottleneck in information search is the fact that some of the texts are written in languages the information seeker does not understand. While this problem cannot be solved completely, some tools and methods can at least be of some help. We suggest two strategies: exploit multilingual gazetteers, nomenclatures, term lists and thesauri for cross-lingual information display, or represent and display the textual information in a language-independent way. These methods can also be used to detect the similarity of documents written in different languages so that users can explore multilingual document collections more conveniently and efficiently. For a more detailed discussion of the methods presented here, see Steinberger et al. (2004).

6.1. Cross-lingual glossing

When information about the extracted term or place name is available in other languages, this information can be displayed inside the text (see Figure 4). For instance, when term lists are available in English and German, English-speaking users can ask the system to identify and highlight the specialist terms found in German text. Additionally, the English translation can be displayed so that the users see in their own language which terms were found in documents written in another language. Figure 4 is an example of automatic detection of place names, and their highlighting and cross-lingual glossing.

6.2. Language-independent representation of textual information

Some textual information can be normalised and represented in a language-neutral way, including date and currency expressions and geographical information. For the representation of references to geographical places, maps and latitude-longitude information are language-neutral (see Figures 1 and 3). Dates written in various formats and in various languages can be represented using a numerical standard (see Ignat et al. 2003). That way, a foreign language text string such as *petnajstega novembra dvatisočpet* can be displayed as 15-11-2005.

The same could be done for currency expressions, telephone numbers, number plates, etc. Such normalised expressions could be highlighted and displayed in the text, similar to the place names in Figure 4.

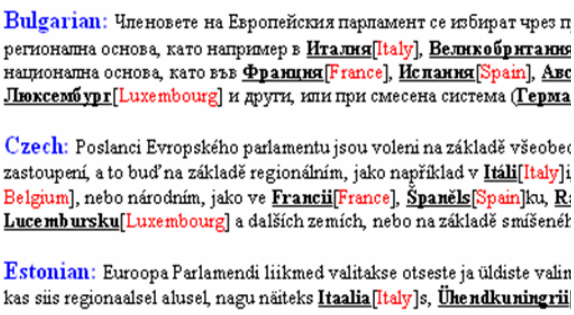

Figure 4. Place names identified and highlighted in Bulgarian, Czech and Estonian texts. Cross-lingual glossing in English.

6.3. Linking similar texts across languages

In news analysis, users are often interested in reading how the same subject or event was discussed by the media in other countries. Therefore, we developed a means to automatically show which clusters in various languages talk about the same story or event. For each given news cluster, the system produces a representation that can be compared to the representations in the other languages. This representation consists of a ranked list of classes from the multilingual thesaurus Eurovoc (Eurovoc 1995), of a ranked list of countries (derived from the country and city names mentioned), of a list of person names automatically identified, and of a list of keywords. The latter is not language-independent, but it benefits from the fact that some names, numbers and cognates are identical even across languages. The complex cluster representation vectors can then be compared so that all clusters being more similar than a certain threshold are identified as being related. For details, see Pouliquen et al. (2004b).

7. Usability status of the software / Future work

The various software tools mentioned in this paper have reached different degrees of maturity. Some of them have been integrated into one complex application so that the results can be browsed in the EMM *News Explorer* (http://press.jrc.it/NewsExplorer), which is currently available in the eight languages English, German, French, Spanish, Italian, Dutch, Slovene and Estonian. Other tools have no interface and need to be combined individually by programmers in order to perform a full analysis of a given text collection. None of the tools is currently set up to be installed on computers outside the JRC. The name recognition patterns are tuned to perform well on news articles, where persons are usually referred to with their title (e.g. *Foreign Minister*) and where persons are often referred to (at least once) by their first and last names. The system in its current state would therefore not work very well on different text types such as scientific publications, where first names and titles are not used.

It is planned to integrate a number of functionalities into a standalone version that can be installed at user sites and that can be used directly by various analysts. This means that automatic document gathering tools like crawlers and search engine result scrapers need to be integrated with document format converters and the whole suite of text analysis tools. Furthermore, an intuitive user interface needs to be prepared, and tools need to be more integrated and robust. We furthermore intend to extend the functionality to more languages and to make the name recognition tools more robust with respect to text types other than news. Over time, we intend to integrate text analysis tools that are not currently integrated (e.g. date recognition) or that will be developed in the future (e.g. recognition of currency expressions). Another plan is to prepare multilingual specialist term lists for various subject areas so that users will be able to select those subject areas for which they want the system to recognise and display terms in their texts, as shown in Figure 1.

8. Acknowledgement

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9. References

Best C, van der Goot E, de Paola M, Garcia T, Horby D. (2002). *Europe Media Monitor – EMM*. JRC Technical Note No. I.02.88, Sept. 2002. http://emm.jrc.org [27.04.2005]

Eisenberg D. (2002). *SVG essentials*. O'Reilly & Associates.

Eurovoc (1995). *Thesaurus Eurovoc - Volume 2: Subject-Oriented Version. Ed.*

3/English Language. Annex to the index of the Official Journal of the EC. Luxembourg, Office for Official Publications of the European Communities. http://europa.eu.int/celex/eurovoc. 1995.

Hammersley B. (2003). *Content syndication with RSS*. O'Reilly & Associates.

Ignat C, Pouliquen B, Ribeiro A, Steinberger R. (2003). *Extending an Information Extraction Tool Set to Central and Eastern European Languages*. In: Proceedings of the International Workshop 'Information Extraction for Slavonic and other Central and Eastern European Languages' (IESL'2003), pp. 33-39. Borovets, Bulgaria, 8 - 9 September 2003.

Kilgarriff Adam (1996). *Which words are particularly characteristic of a text? A survey of statistical approaches*. Proceedings of the AISB Workshop on Language Engineering for Document Analysis and Recognition.

Pouliquen B, Steinberger R, Ignat C, de Groeve T. (2004a). *Geographical Information Recognition and Visualisation in Texts Written in Various Languages*. In: Proceedings of the 19th Annual ACM Symposium on Applied Computing (SAC'2004), Special Track on Information Access and Retrieval (SAC-IAR), vol. 2, pp. 1051-1058. Nicosia, Cyprus, 14 - 17 March 2004.

Pouliquen B, Steinberger R, Ignat C, Käsper E & Temnikova I. (2004b). *Multilingual and Cross-lingual News Topic Tracking*. In: CoLing'2004 proceedings, Vol. II, pp. 959-965. Geneva, August 2004.

Pouliquen B, Steinberger R, Ignat C, Zaghouani W, Žižka J. (2005). *Detection of Person Names and their Translations in Multilingual News*. Colloque 'Traitement lexicographique des noms propres'. Tours, France, 24 March 2005.

Steinberger R, Pouliquen B, Ignat C. (2004). *Exploiting Multilingual Nomenclatures and Language-Independent Text Features as an Interlingua for Cross-lingual Text Analysis Applications*. In: Information Society 2004 (IS'2004) - Proceedings B of the 7th International Multiconference - Language Technologies, Ljubljana, Slovenia.

Session 10

Open sources treatment

Use of ASTER Thermal Infrared Image Data for Nuclear Safeguards Purposes

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Abstract:

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 - in case the atmospheric conditions and other influencing factors are known. Moreover, thermal data could be used to evaluate whether significant changes have taken place in the thermal characteristics of these materials over time.
Thus, the use of thermal infrared imagery for the monitoring of (heat generating) nuclear facilities 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. Compared to LANDSAT 7 owning one thermal channel in the range of 10.40 to 12.50 µm, Aster offers five channels between 8.125 and 11.65 µm. Besides the (registered) radiance at sensor TIR (Level 1A and 1B), so-called on-demand products can be ordered free of charge for the archive imagery, i.e. brightness temperature at sensor (AST_04), surface emissivity (AST_05), surface kinetic temperature (AST_08), surface radiance TIR (AST_09T).
Some case studies using ASTER on-demand products will illustrate the potential of thermal infrared data for the safeguards-related monitoring of nuclear facilities.

Keywords: thermal infrared, ASTER imagery, nuclear safeguards purposes

1. Introduction – "Think thermally"

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.

Thermal imagery can not be interpreted in the way panchromatic or multispectral aerial or satellite imagery is usually interpreted for safeguards purposes or any other application. In order to analyse and interpret thermal image data, the user has to develop a "thermal understanding" (see Jensen [1]) on:

- The interaction between the energy from the sun or from the earth with the atmosphere;
- the interaction between the energy from the sun or from the earth with the various terrain components;
- the sensor in terms of measuring the terrain's emitted thermal infrared electromagnetic radiation;
- the appearance of noise due to the sensor system or the terrain characteristics.

Compared to the analysis of "normal" optical remote sensing imagery, the user has to consider several aspects regarding thermal imagery in order to avoid misinterpretations. Two buildings being next to

one another on the ground with the same kinetic temperature could have different apparent temperatures due to different emissivity. Emissivity is known as the ratio between the radiant flux exiting a real-world radiating body and a blackbody. The emissivity of an object may be influenced by a number of factors, including material parameters (colour, mineral composition, surface texture, density, porosity resp. pore volume, permeability, moisture content, heat sources inside) and environmental parameters (topographic position in the field, orientation of the surface to the sun, meteorological conditions, microclimate, humidity, time of day, season). The material and environmental parameters act simultaneously and independent as well as in the same and opposite direction. Moreover, the environmental parameters are difficult to control and not reproducible. For a comprehensive overview on the physical basics of thermal infrared and the thermal radiation laws please see Jensen [1], NASA' Tutorial [2], Quattrochi [3] and Rees [4].

2. Availability of satellite thermal infrared data

At the present time, only the LANDSAT and ASTER (Advanced Spaceborne Thermal Emission and Reflection Radiometer) satellites offer image data from the thermal infrared region. 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. 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 (See Figure 1) .ASTER, launched in 1999 too, features the thermal subsystem TIR with five channels 8.125 and 11.65 µm at a spatial resolution of 90m and a temperature accuracy of 0.3°K (see Figure 1 and Table 1). More information on LANDSAT 7 ETM+ and ASTER are given in Abrams et al. [5] and Kramer [6].

ASTER image data are available in world-wide coverage via the virtual EOS Data Gateway[1]. Level-1A (AST_L1A) and Level-1B (AST_L1B) data are offered at low prices. ASTER L1A data are formally defined as reconstructed, unprocessed instrument data at full resolution. They consist of the image data, the radiometric coefficients, the geometric coefficients and other auxiliary data without applying the coefficients to the image data, thus maintaining original data values. The L1B data are generated by applying these coefficients for radiometric calibration and geometric resampling. Besides, higher-level (Level-2) ASTER products can be ordered free of charge on-demand for the given ASTER-L1B image data, i.e. [5]:

- Brightness Temperature at the Sensor (AST_04), for ASTER's five thermal-infrared bands (8-12 µm, bands 10-14). Brightness temperature is the apparent observed temperature, assuming a surface emissivity of 1.0 (i.e., as if the object were a blackbody). The calculations are performed starting with the radiance at sensor as input; no atmospheric correction is included for this product.
- Emissivity (AST_05), contains surface emissivity at 90-m resolution generated only over the land from ASTER's five thermal infrared channels. Surface emissivity is required to derive land surface temperature (AST08) data, also at a resolution of 90 meters.
- Surface Reflectance/VNIR & SWIR (AST_07), provides surface reflectance for each of the nine VNIR and SWIR bands at 15-m and 30-m resolutions, respectively. The results are obtained by applying an atmospheric correction to radiances reported by the ASTER sensor. The atmospheric correction removes effects due to changes in satellite-sun geometry and atmospheric conditions.
- Surface Kinetic Temperature (AST_08), offers surface temperatures at 90-m resolution generated only over the land from ASTER's five thermal infrared channels. Land surface temperatures are determined from Planck's Law, using the emissivities from AST_05 to scale the measured radiances after correction for atmospheric effects.
- Surface Radiance/VNIR & SWIR) (AST_09), includes surface radiance for each of the nine VNIR and SWIR bands at 15-m and 30-m resolutions, respectively. The results are obtained by applying an atmospheric correction to radiances reported by the ASTER sensor. The atmospheric correction removes effects due to changes in satellite-sun geometry and atmospheric conditions
- Surface Radiance/TIR (AST_09T), provides surface leaving radiance, in W m^{-2} sr^{-1} μm^{-1}, for the five ASTER TIR channels at 90 m spatial resolution. In addition, the down welling sky irradiance in W m^{-2} sr^{-1} μm^{-1} for the five ASTER TIR channels is also provided. Atmospheric correction has been applied and the surface leaving radiance is valid for the clear sky portion of scenes. This radiance includes both surface emitted and surface reflected components.

[1] http://edcimswww.cr.usgs.gov/pub/imswelcome

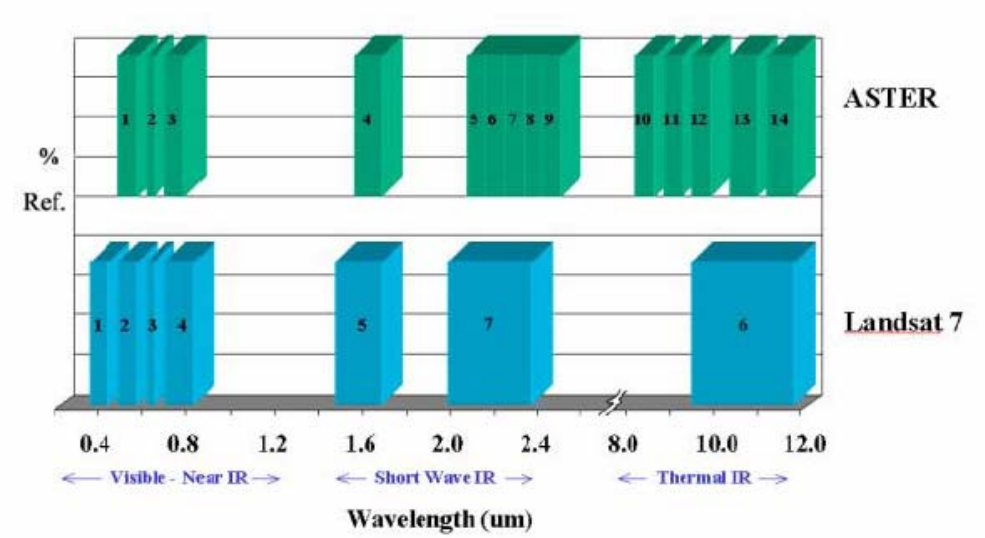

Figure 1: Comparison of the spectral bands between ASTER and Landsat-7 ETM+ (Abrams et al. [5])

Subsystem	Band No.	Spectral Range (μm)	Spatial Resolution, m	Quantization Levels
VNIR	1	0.52-0.60	15	8 bits
	2	0.63-0.69		
	3N	0.78-0.86		
	3B	0.78-0.86		
SWIR	4	1.60-1.70	30	8 bits
	5	2.145-2.185		
	6	2.185-2.225		
	7	2.235-2.285		
	8	2.295-2.365		
	9	2.360-2.430		
TIR	10	8.125-8.475	90	12 bits
	11	8.475-8.825		
	12	8.925-9.275		
	13	10.25-10.95		
	14	10.95-11.65		

Table 1 : Characteristics of the 3 ASTER sensor systems (Abrams et al.[5])

3. Case studies using ASTER thermal infrared bands

Case studies were carried out for some (heat generating) nuclear facilities. First of all, we compared the on-demand products AST_5 and AST_08, estimated by the ASTER temperature/emissivity separation algorithm with the values for temperature and emissivity computed by ENVI on the basis of the thermal infrared Level 1A/1B data (and ASTER_09T respectively, if Level 1A/1B were not available). The steps by ENVI included atmospheric correction by regression pixel and the calculation of emissivity (and temperature) by the emissivity normalization technique. Preliminary results indicate that AST_05 and AST_08 seem to have a wider variance of values: for relatively small values, the ENVI-based values are slightly bigger, for relatively high values, the ENVI estimates smaller values.

In the following, AST_08 data were analysed for the enrichment and conversion facilities at Trciastin/Pierrelatte, France, the reprocessing plant in La Hague, France and the development of the enrichment facility at Natanz, Iran. Figure 2 shows the surface kinetic temperature (right) for the nuclear installations at Tricastin/ Pierrelatte compared to the optical/near infrared data (left) at April 24, 2004 at 10:41 am. All water bodies have smaller temperatures than the other surfaces, but also the four “cascade” buildings (gaseous diffusion) indicate a relative low surface temperature. The other facilities (for gaseous diffusion, conversion, fuel fabrication, waste management, laboratories and enrichment pilots) at this location have relative high surfaces temperatures, especially the system of

four pressurized water reactors for electricity production. Moreover, there is a slight hint for a hot water plume from the reactor complex. Figure 3 illustrates the dependency on the season.

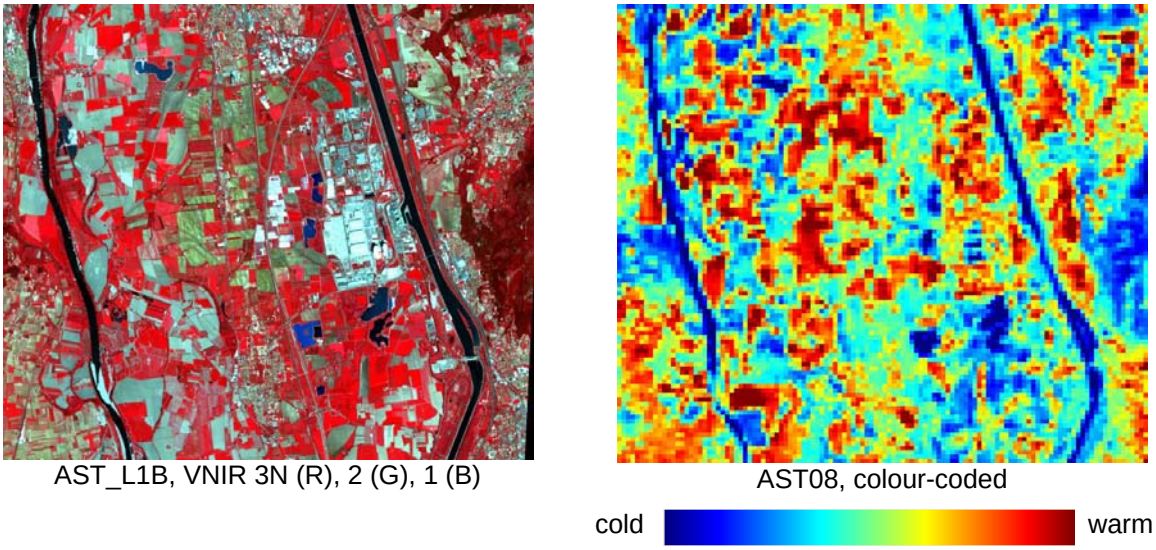

Figure 2: Uranium conversion and enrichment installations at Tricastin/Pierrelatte, France by optical/near infrared (left) and thermal infrared (right) data, acquired at April 24, 2004, 10:41 am

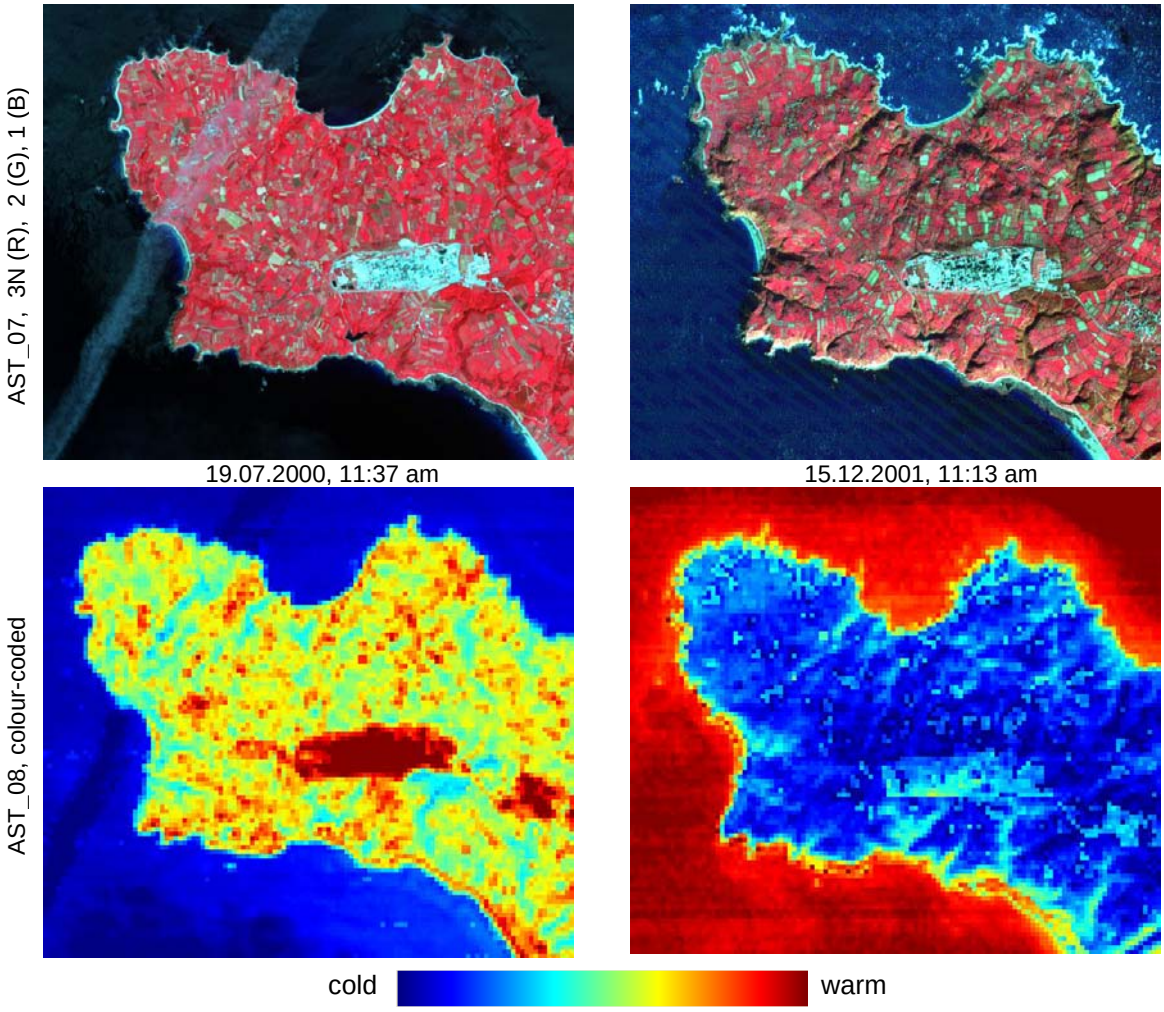

Figure 3: Comparison of relative summer and winter temperature for the preprocessing plant at La Hague; France by optical/near infrared (top) and thermal infrared (bottom) data

For the reprocessing plant at La Hague, France, the surface kinetic temperature in July 2000 and December 2001 at almost the same daytime were compared. 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.

Finally, the use of thermal infrared data for the monitoring of nuclear facilities under construction is presented in Figure 4. The development of Natanz nuclear enrichment facility is shown by optical/near infrared data and the kinetic surface temperature from July 2000 to May 2001 and November 2001. In comparison to the dark sandy deserts in the direct neighbourhood, the man-made materials of the facility indicate a much lower surface kinetic temperature. As the installations increased during the construction process, also this area of relative small temperatures expanded.

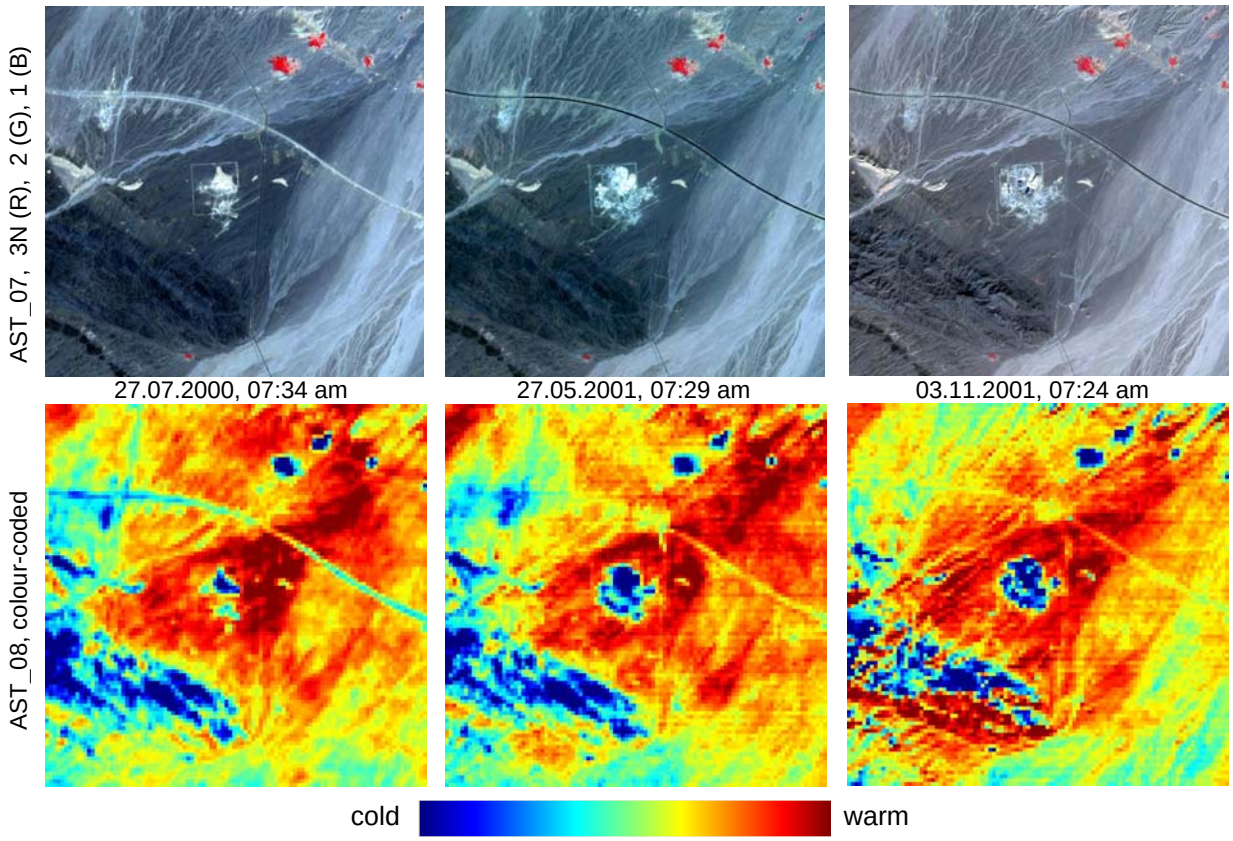

Figure 4: Development of the Iranian Natanz site between July 2000 and November 2001, shown by optical/near infrared (top) and thermal infrared (bottom) data

4. Conclusions and Outlook

The use of thermal infrared imagery for the monitoring of (heat generating) nuclear facilities seems to be reasonable. The image data given by the ASTER thermal infrared system, i.e. provided with the on-demand product AST_08 (surface kinetic temperature) 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. Thermal infrared data gathered by future commercial satellites with better spatial resolution could be used to verify the member state's declarations, to monitor the construction of facilities and to detect undeclared activities.

The description of an object as "cold" or "warm" compared to its surroundings has to be given in connection with the material and environmental parameters. Environmental influences may overlay or modify the natural or artificial thermal radiation or even result in thermal anomalies. The thermal

behaviour of two different materials during the course of the day shows, that general statements in terms like “warmer or colder than another object” are not possible.

The next steps will include the analysis of nighttime imagery that is also available as ASTER on-demand product. A comprehensive processing of all available ASTER image bands (including VNIR and SWIR daytime data, TIR daytime and nighttime data) will be carried out by data fusion and combined analysis. Moreover, advanced techniques for the detection of thermal anomalies, as introduced by Schäfer [7] and for the classification of the five different thermal bands, as proposed by Franck [8] will be evaluated. The study should be expanded to other types of nuclear facilities.

5. References

[1] Jensen, J.R., *Remote Sensing of the Environment. An Earth Resource Perspective.* Chapter 8: Thermal Infrared Remote Sensing, Prentice Hall, Upper Saddle River, 2000. pp. 243-283

[2] NASA, *NASA’s Remote Sensing Tutorial, Section 9, The Warm Earth: Thermal Remote Sensing* (Download: http://rst.gsfc.nasa.gov)

[3] Quattrochi, D.A. & J.C. Luvall (ed.), *Thermal Remote Sensing in Land Surface Processes*, CRC Press, Boca Raton, 2003

[4] Rees, W.G., *Physical Principles of Remote Sensing*, 2nd edition, Cambridge University Press, Cambridge, 2001

[5] Abrams, M.; Hook, S. & B. Ramachandran, *ASTER User handbook*, Version 2, 2002 (Download: http://asterweb.jpl.nasa.gov/documents/aster_guide-v2.pdf)

[6] Kramer, H.J., *Observation of the Earth and Its Environment, Survey of Missions and Sensors,* 4th edition. Springer, Berlin, 2002

[7] Schäfer, M., *Auswertung thermaler Informationen von LANDSAT-7 ETM+ und ASTER-Daten zur Erfassung von Landoberflächentemperaturen*. Unpublished Diploma Thesis, TU Clausthal, 2003

[8] Franck, R., *Thematic Classification and Determination of Surface Parameters from Airborne Multitemporal Imagery*, Logos-Verlag, Hamburg, 2001

Classifying nuclear power plant facilities using an already developed "key"

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Abstract

With the aid of aerial photographs and satellite imageries, typical features of pressurised water and boiling water reactors were determined. Among a number of other characteristics, it was found that, in satellite images, there are features that could be used to distinguish between a fossil fuelled power plant and a nuclear reactor. From such imageries it is also possible to recognise a civil reactor from that used for defence purposes. Moreover, it was found that from satellite imageries it is possible to distinguish between a PWR and a BWR. The Software eCognition, developed by Definiens Imaging GmbH, was programmed to recognise the features developed earlier. This was then validated using high-resolution (0.61m to 1m) imagery. This paper describes how the software could automatically locate and identify a nuclear reactor facility in a large satellite scene and some of the preliminary results are given. The work is carried out within the European Union's Global Monitoring for Security and Stability (GMOSS) project.

Keywords: satellite imagery; automatic feature recognition; ontology

1. Introduction

Iran signed the NPT on 1 July 1968 and deposited the instrument of ratification on 2 February 1970 with the USA, on 10 February 1970 with the USSR, and on 5 March 1970 with the UK. It signed the safeguards agreement on 15 May 1974 (INFCIRC 214) and the Additional Protocol on 18 December 2003. It acts as if it is in force until it has ratified the Protocol. In spite of this, Iran's compliance with the NPT and the Safeguards agreements came into question.

In order to defuse the political tensions, the European Union (EU) began to make considerable efforts to persuade Iran to halt its enrichment and reprocessing activities. France, Germany and the UK led the discussions with Iran. In view of EU's involvement, the members of the recently established Global Monitoring for Security and Stability (GMOSS) project under the European Commission's (EC) Network of Excellence felt that as a test case it will study Iran's nuclear activities. Monitoring arms control treaties using commercial remote sensing satellite imageries forms an important part of the GMOSS project.

In this paper, Iran's nuclear programme is briefly described. Then, using already established "key" for certain types of nuclear reactors, an attempt has be made to evolve a method of detecting such facilities automatically using computers. A software, eCognition, developed for object oriented image analysis has been applied in this study.

2. Nuclear Activities

Uranium mining

Systematic exploration began in 1974. It is estimated that the resource exceeds 5,000 tonnes, but may be more. The Saghand (32° 32'N, 55 ° 15'E) uranium deposit contains up to 1,550,000 tonnes of ore

Milling

The Yazad facility was established in about 1989. Two additional sites are at Bandar Abbas and Bandar-e-LangehDefence related laboratory scale facility is at Tehran Nuclear Research Center (TNRC).

Conversion facilities

Possible locations are at Isfahan Nuclear Technology Center (ENTC) and Rudan Nuclear Research Center (1988) near Fasa (28° 56'N, 53° 39'E).

Fuel fabrication facility

Nuclear Fuel Research & Production Center (NFRPC), located at Isfahan, was founded in 1974. Present location is SE of Isfahan at Roshandasht.

Heavy Water Production Plant

A heavy water production plant is under construction at Arak.

Reactors

A 5MWth research reactor was built in the 1960s at Tehran Research Center (TRC). It uses low-enriched uranium. Other facilities are a Miniature Neutron Source Reactor, a 30kWth, a Heavy Water Zero Power Reactor, a Graphite Sub-Critical Reactor and a Light Water Sub-critical Reactor all are located at Isfahan.A Light Water Power Reactor 1,000MWe is under construction at Bushehr.

Enrichment

A number of techniques are being explored. For example, an **Electromagnetic Isotopic Separation (EMIS)** facility (desktop equipment provided by China in 1987) is located at Esfahan. Another facility is at Karaj. A Laser facility (TNRC) houses a Laser Research Center since 1992. Under its **Centrifuge enrichment programme**, Iran is developing a pilot and a commercial centrifuge plant at Natanz. As of February 2003, over 100 of about 1,000 planned centrifuge casing had been installed at the pilot plant. The IAEA was informed that the pilot plant will begin test runs with smaller number of centrifuges in June 2003. The commercial facility housing over 50,000 centrifuges is scheduled to start in early 2005. Two other facilities are reported at Lashkar-Abad nera Hashtgerd and near Ramandeh village.

3. Description of object oriented Image analysis

Detecting nuclear facilities and their characteristic features using methods of object oriented image analysis is possible in principle [1]. Nevertheless, automating the detection of nuclear facilities and their characteristic features therefore needs knowledge about the features to detect, the remote sensing data used and the pattern how these features are imaged in the data used. This knowledge has to be formulated in a computer understandable way. Methodologies to formulate this knowledge, i.e. to make a computer "understand" an image are manifold. The most important ones are:

Semantic nets

Objects are described by their semantic meanings, i.e. by their interrelationships to other (classified) objects.

Ontologies

Ontologies try to describe objects occurring in real world and their meaning for humans in general. When describing the objects, their functionality and their relationships to other objects are described, if applicable. There are different basic techniques possible, depending on the objects to describe, e.g. simple taxonomies or descriptions as like in dictionaries. A very simple ontology for nuclear power plant would look like the following: *a power plant which uses nuclear energy to generate electricity.* There are the key words *power plant* and *nuclear energy* in the description. These keywords must be described well in order to "know" what is meant by *nuclear power plant*. The description of *power plant* then should contain the key word *facilities* and maybe *cooling tower* or something else typical for power plants. An ideal feature detection system then would be able to translate the verbal description of the typical facilities into a visual description. Until recently, this has been done manually.

Fuzzy Logic

With fuzzy logic in general it is possible to describe uncertainties of human knowledge. Fuzzy classifications can be best applied in cases, where it is not absolutely clear whether the objects to classify belong to one class or to another class (which is actually valid for all kind of objects). Typical cases are (costal) transition zones in images. Usually one can outline the (coast) line in a sharp manner, but there is always a certain degree of uncertainty if the drawn line clearly delineates the (coast) line. Thus, using degrees of membership to more then one class for classified objects expresses the reliability of a classification result (e.g. sea or land).

Object Oriented Programming Languages

They can be seen as the computer understandable translation of ontologies. In fact methods like UML (unified modelling language) are a pre-step before generating the software code. Mechanisms, such as inheritance accelerate the programming task and simultaneously reflect the human way of thinking in categories and sub-categories.

eCognition as software for object oriented image analysis integrates some of these methodologies [3]. For example, class descriptions are all formulated in a fuzzy manner; relationships to objects of other classes can be formulated and used for classification and mechanisms of inheritance can be used.

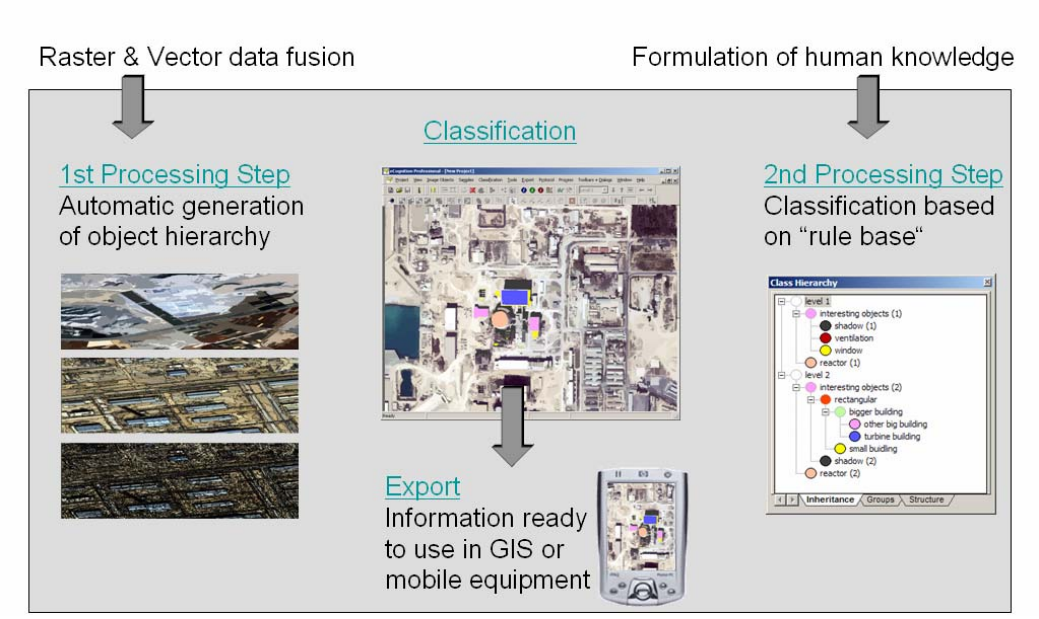

Figure 1: Workflow Object Oriented Image Analysis

4. Key features of power plants and their relations

Some of the key features of nuclear power plants that were described by Jasani [2] and their relations are provisionally listed in the table below. The overview describes some of their characteristics and differences to fossil fuelled power plants. The focus is on relations between single components and common knowledge about the features in remote sensing imagery.

Power plant types and their components	**Description, characteristics and differences to similar features**	**What do we know about this feature in Remote sensing imagery**
nuclear reactor types		
pressurised nuclear power reactor (PWR)	coolant is isolated from the primary steam generator and the turbine	
boiling water nuclear power reactor (BWR)		
gas cooled reactors (GCR)		
advanced gas cooled reactors (AGR)		
pressurised heavy water reactor (PHWR)	coolant is isolated from the primary steam generator and the turbine	
light water reactor LWR		
major components of a nuclear reactor (PWR)		
Reactor	in containment building	Large round building, close to large rectangular buildings
steam generator	in containment building	
fuel storage	in containment building	
fuel building	to store new fuel and spent fuel in a cooling pond	
Reactor control building		

turbine and electricity generator building		large rectangular building with windows and ventilation exhausts on top
diesel generator		
switchyard		
cooling tower		round towers or rectangular shape, dependent on sun angle large shadow
hybrid cooling tower		
mechanical draft cooling towers		
large water source	river, lake, water basin	mostly dark homogeneous areas, near infrared channel
warm water outlet		thermal channels - higher values
exhaust stacks		characteristic shadow (sunangle dependent)
whole facility surrounded by dominant wall/fence		narrow, linear structure
civil purpose		
defense purpose		
detect undeclared nuclear activities		change detection close to existing facilities, trying to hide changes (camouflage, underground activities)
fossil fuelled power plant	same basic principles as nuclear: heat generated by consuption of fuels, produce steam, drive electric generators; no cylindrical buildings similar to those for PWRs	
fuel storage	heaps of coal, cylindrical tanks (e.g. 60 m diameter) for oil and gas; fuel is stored outside at the site of the plant	
heaps of coal		
oil and gas tanks	e.g. about 60 m diameter	
steam generator		
electric generator		
rectangular shaped buildings		
exhaust stacks	tall, thin, can be located on top of generator building (= difference to nuclear)	
railway network	extensive for solid fuel to transport large quantities to power plants	
conveyor belt system/network	for coal fuel	
whole facility surrounded by wall/fence		

Table 1: Key features of power plants and their relations

5. Application of the key describing nuclear power plants

Here a methodology to describe expert knowledge about nuclear power plants and facilities in a way that the knowledge is usable for feature recognition in images is presented. The outcome can be understood as a semantic net for describing the ontology of features typical for nuclear power plants. Thereby, an exemplary description was created using eCognitions class hierarchy editor. This way, it is possible to use concepts like inheritance and fuzzy-logic. The used image objects are extracted semi-automatically using Multiresolution Segmentation and manual fusion of selected object.
As exemplary data a Quickbird image of Busheer (Iran), taken on 28th February 2004, was used. The image shows a pressure water reactor (PWR) which is under construction. Since one reactor dome and most of its surrounding buildings are already finished, the ontology for this typical pattern could be developed and applied to classify the semi-automatically generated image objects. Thereby, the knowledge description was based upon B. Jasani's article which describes typical patterns of different types of nuclear power plants in Germany. The description of the buildings which are typical for PWRs is done firstly based on shape – namely size and roundness (for the reactor) respectively rectangularity (for the turbine and control building). In order to focus on relevant buildings, only objects within a buffer of 120 pixels up to 200 pixels around the reactor building were taken into account, where a distance of 120 pixels leads to a fuzzy membership degree of 1.0 and a distance of 200 gives back a fuzzy membership degree of 0.0 (see Figure 1). Secondly, spatial contextual information was taken into account in order to identify the buildings. As such the turbine building was identified according to the existence of its characteristic windows and ventilation exhausts on top (Figure 2, Figure 3).

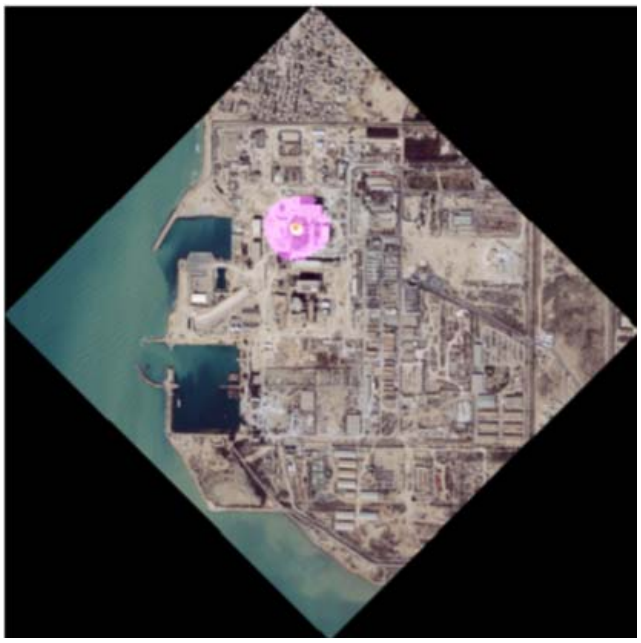

Figure 1: Quickbird image from Busheer. Orange: reaktor building. Pink: search buffer for interesting buildings.

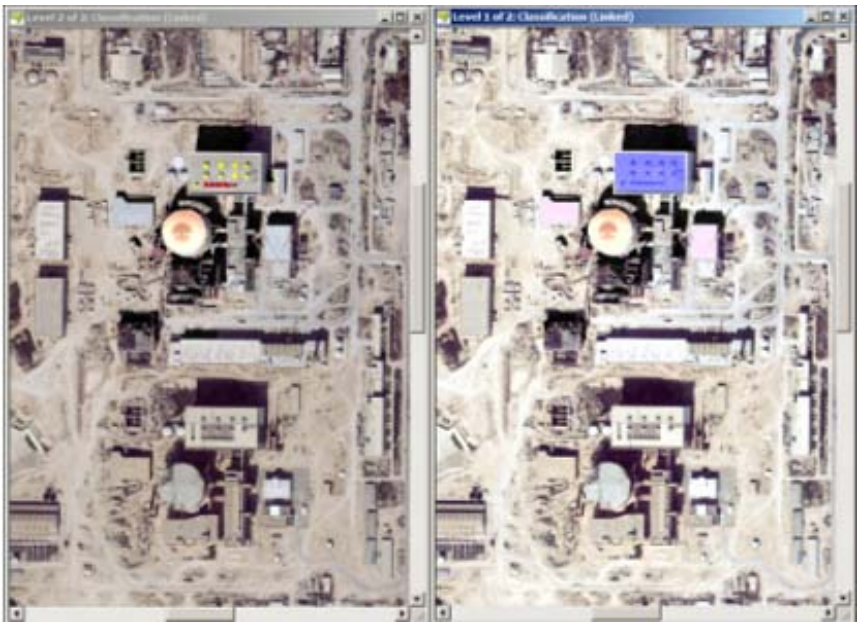

Figure 2: Left: Classified windows (yellow) and ventilation exhausts (red) on top of the identified turbine building (right). The turbine building is the only bigger building close to the reactor with ventilation exhausts and windows on top.

Figure 3: Class description of „turbine building“. The turbine building is close to the reactor (Distance to), a bigger (Area) and rectangular (Rectangular fit) building and has windows and ventilation exhausts on top (Existence of ventilation & Existence of window).

6. Conclusions and Outlook

The analysis shows that parts of the key could be formulated successfully by using eCognition's rule base. The transfer to other sites is planned within GMOSS project.

Current limitation is that image objects that result from one segmentation level do not always represent objects of interest especially of different scales. Here an automated approach for iterative segmentation is currently tested and looks very promising.

The key for research facilities is now being developed and the application in eCognition will be tested in the near future.

7. References

[1] Canty, M.J., Niemeyer, I., Richter, B. and G. Stein (1999): *Wide-area Change Detection: The Use of Multitemporal Landsat Images for Nuclear Safeguards.* Journal of Nuclear Materials Management (JNMM), Vol. XXVII, No. 2, 1999, 19-24

Jasani, B., Canty, M.J., Niemeyer, I., Richter, B., Stein, G. and D. Klaus, *Possible Applications of Commercial Satellite Imagery for International Safeguards: Some Case Studies using Optical and Radar Data*. Berichte des Forschungszentrum Jülich 3650, Jülich, 1999

Jasani, B., Canty, M.J., Richter, B., Stein, G., Niemeyer, I. and M.D. Ward *Commercial Satellite Imagery and Safeguards: Some Case Studies Using Multispectral and Radar Data*. An Executive Summary. SRDP-R266, JOPAG/05.99-PRG-293, Task A.20/JNT D00988, 1999

Klaus, D., Canty, M.J., Poth, A., Voss, M., Niemeyer, I. and G. Stein, *Satellitenbildauswertung mit künstlichen Neuronalen Netzen zur Umweltüberwachung. Vergleichende Bewertung konventioneller und Neuronaler Netzwerkalgorithmen und Entwicklung eines integrierten Verfahrens.* Schriftenreihe des Forschungszentrum Jülich, Reihe Umwelt/Environment, Band 15, Jülich, 1999

Niemeyer, I., Canty, M.J. and M. Baatz, *Fractal-Hierarchical Pattern Recognition for Safeguards Purposes.* Proc. of the 2nd International Symposium on Operationalization of Remote Sensing, ITC, Enschede, 16-20 August 1999. Enschede (CD-Rom), 1999

[2] Jasani, B; *Identification of key features of nuclear reactors for interpretation of imageries from remote sensing satellites*", Journal of Nuclear Materials Management, Spring 2004, Volume XXXII, No. 3, pp. 28-36.

[3] Definiens Imaging; *eCognition User Guide*, Munich, Definiens Imaging GmbH, 2004, http://www.definiens-imaging.com

Evaluation of remote sensor systems for monitoring uranium mines

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Abstract:

The use of satellite remote sensing instrumentation to gather multi-spectral and hyper-spectral data is often viewed as the panacea for monitoring activities. This paper provides an overview of these technologies and data, and their applicability to the problem of uranium production monitoring. In this paper, we discuss the available instruments, their parameters and limitations, on-going work in this area, and potential observables in uranium mining and milling operations as they pertain to the international safeguards community.

Keywords: satellite imagery; multi-spectral; hyper-spectral; remote sensing; uranium mining; Additional Protocol

1. Introduction

Uranium deposits are found throughout the world with the largest deposits located in Australia (28%), Kazakhstan (20%), Canada (14%) and South Africa (10%) [1]. As these and other states sign-up to the Additional Protocol to their Safeguards Agreements with the International Atomic Energy Agency (IAEA), they will need to declare a broader scope of their nuclear activities, including uranium mining activities. This could significantly increase the burden on IAEA resources for safeguards implementation at uranium mines in terms of inspector time and monitoring technology deployed.

There has been interest expressed to use multi-spectral and hyper-spectral sensors for remote verification of uranium mining and milling. The use of these techniques by international monitoring communities has been seen as a way to minimize costs while improving inspection performance. Providing large area coverage, they can potentially minimize, alleviate, or eliminate the need for inspectors at the mining site. However, there are many issues associated with analyzing remotely sensed data to quantitative results. The interactions of photons with the atmosphere and the targets of interest, whether in the solid, liquid, or gas phase, can be complex, significantly impacting the resultant data. Effective interpretation requires analysts knowledgeable and trained in solar radiation, atmospheric physics, optics and detectors, and multi-spectral and hyper-spectral data processing algorithms. This paper examines, in general terms, issues associated with using remote spectral tools, such as multi-spectral or hyper-spectral satellite platforms, for monitoring activities. It will also examine one area of particular interest to the international nonproliferation community: uranium mining and milling operations.

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2. Sensor principles and background

2.1. Spectroscopy to imaging

Spectroscopy has been used in the laboratory by physicists and chemists for over 100 years to record information directly related to the chemical bonds in molecules. By measuring the specific wavelengths of radiation that are absorbed by a sample, a unique “fingerprint” of the material is obtained. Defensible in a court of law, this information is used extensively in forensic activities by crime agencies across the globe.

A spectrometer records the photons in the instrument field of view that strike the detector. The photons are collected at particular wavelengths using some form of dispersive element or filter mechanism. An imaging spectrometer expands this concept by using an array of detectors. The imaging spectrometer captures photons from each subregion in a larger area onto individual detectors, resulting in an image of the area. This approach, which provides material interaction information about different areas on a sample, has a number of data tradeoffs that will be discussed later. In a laboratory, sample conditions and the instrument environment can be tightly controlled. This is critical because light interacts with all matter, in absorption, reflection, and emission modes. The interacting media can be the sample of interest, the environment around the sample, or changes in the instrument itself. Advances in technology have improved the stability of instruments, allowing them to be used outside the laboratory. But the above issues remain.

Applications benefited from technology developments, and robust systems were identified to be flown on airborne platforms to study wide geographic areas. The first imaging systems to be deployed on airborne platforms utilized broad spectral bands in the visible range. These systems, known as panchromatic instruments, result in black and white images, i.e. photographs, of the sample area. Generally these images do not contain radiometric information. Analysis of these images is performed visually for shape and contrast content.

The second-generation instruments were known as multi-spectral with a few bands ranging from 0.4 to 1.1 μm (four bands) and a thermal band at 10.4-12.6 μm. The third generation satellites, Landsat-4 to Landsat-7, carried sensors with seven spectral bands ranging from 0.45 to 0.9 μm (four bands), 1.55-1.75 μm (band five), 2.08-2.35 μm (band six) and a thermal band with 10.4-12.5 μm wavelengths. The Landsat-7 sensor had an additional panchromatic band (0.50-0.90 μm). Table 2, which will be discussed in more detail later, lists technical information regarding the Landsat satellites. A fourth generation instrument was placed on board the ASTER satellite with 14 spectral bands, Table 3. These instruments provide general classes of chemical information. Hyper-spectral instruments followed multi-spectral systems. Moved out of the lab and onto airborne platforms in the mid 1980’s, the commercial systems typically have at least 220 bands, with wavelengths ranging from 0.40 to 2.50 μm.

2.2. Spectral component to imaging

As stated earlier, when moving from the controlled environment of a laboratory to the field, complex interactions of light with matter will manifest themselves in the data. The farther the sample is from the detector, the greater the impact of these complexities. This became obvious as imaging systems were placed on airborne and satellite platforms miles away from the material they were analyzing. In the laboratory, the photons incident upon a sample, and the collection geometry of the sample, is controlled by the experimentalist. The spatial extent of the material of interest is controlled to provide optimum sensor fill and geometry. This is not possible with the large sample area being investigated using platform-based imaging systems. Furthermore, in the laboratory, the source is fully characterized and can readily be subtracted from the data, leaving only the signature of the compound of interest. For the passive imaging systems of interest here, the source is the sun for daytime collections and self-emission of the target for nighttime imaging. To obtain the material features, this variable source must be removed from the data, leaving only the signature of the interaction.

The intervening atmosphere also provides significant interaction. Because of the complexity of all the interactions of the source prior to reaching the sensor, there is significant signal preprocessing that is necessary to extract a ground signature. This is done using one of various available models. To

improve the accuracy of the results, these models require inputs that define current atmospheric conditions such as temperature, pressure, aerosol density and size distribution.

For remote imaging of solids, the primary property of interest is reflectance. Individual materials reflect the light in a very specific way as a function of wavelength that provides a unique signature for most materials. Typically satellite and airborne systems are calibrated to report radiance (Watts/m^2/μm/sr) that is incident on the detector.

For completeness, we have included the governing equation for *at sensor radiance*, a term commonly used in the imaging community, Equation 1.

$$L_\lambda = \{E_{S\lambda}\cos(\sigma)\tau_1(\lambda)\frac{r(\lambda)}{\pi} + \varepsilon(\lambda)L_{T\lambda} + F[E_{ds\lambda} + E_{d\varepsilon\lambda}]\frac{r_d(\lambda)}{\pi} + (1-F)[L_{bs\lambda} + L_{b\varepsilon\lambda}]r_d(\lambda)\}\tau_2(\lambda) + L_{us\lambda} + L_{u\varepsilon\lambda} \quad (1)$$

Each of these terms has a different contribution to the final measurement. This equation includes the solar irradiance (E) modified by the solar angle, transmission and distance, the radiance (L) of the object, which is modified by the object emissivity, and the scattering (F) and absorption terms from the source to the object and the object to the detector. This equation cannot be solved directly and for illustration purposes it is more instructive to look at Equation 2, which identifies each component shown in
Figure 1, where L is radiance and subscript meanings are shown in Table 1.

$$L = L_A + L_D + L_B + L_E + L_G + L_H + L_C + L_F \quad (2)$$

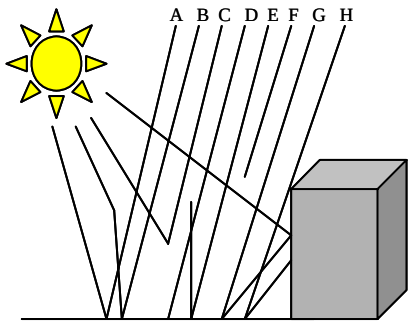

Figure 1: Contributions to the signal at the sensor (located where letters are).

Table 1: Photon interactions in remotely sensed data.

Subscript	Meaning
A	Photons that interact with target
B	Photons that interact with target and atmosphere
C	Photons that interact with atmosphere only
D	Photons that are absorbed by target
E	Photons that are absorbed by atmosphere after interacting with target
F	Photons that are absorbed by atmosphere before interacting with target
G	Photons that interact with target and are scattered to interact with other materials prior to reaching sensor
H	Photons that interact with target and are scattered and absorbed by other material

From this table, it is seen that there are many complex interactions, some of which reach the detector and some that do not.

The radiance measurement at the sensor itself does not directly provide information about the material of interest. The data collected by multi-spectral and hyper-spectral sensors generally are processed by the analyst into reflectance units, allowing for mathematical manipulation and direct comparison to pure material obtained in the laboratory. However, this processing of the image data to reveal ground reflectance is not trivial. A thorough understanding of solar radiation and its interaction with atmospheric constituents is needed, along with the exact geometry of the collection. Accurate detection and identification of materials is dependent on the spectral coverage, spectral resolution, signal-to-noise of the spectrometer, the abundance of the material and the strength of absorption features for that material in the wavelength region measured.

Figure 2 is an illustration of both multi-spectral data, typically identified as five to twenty moderately wide spectral bands, and hyper-spectral data, identified as hundreds of narrow spectral bands with a bandwidth of approximately 10 nm. These plots demonstrate the relative data differences of the two techniques. Intended to illustrate the distinct signature or spectral response, at about 0.68 μm, caused by chlorophyll and commonly referred to as the "red edge", this figure also shows the more detailed chemical information contained in hyper-spectral data.

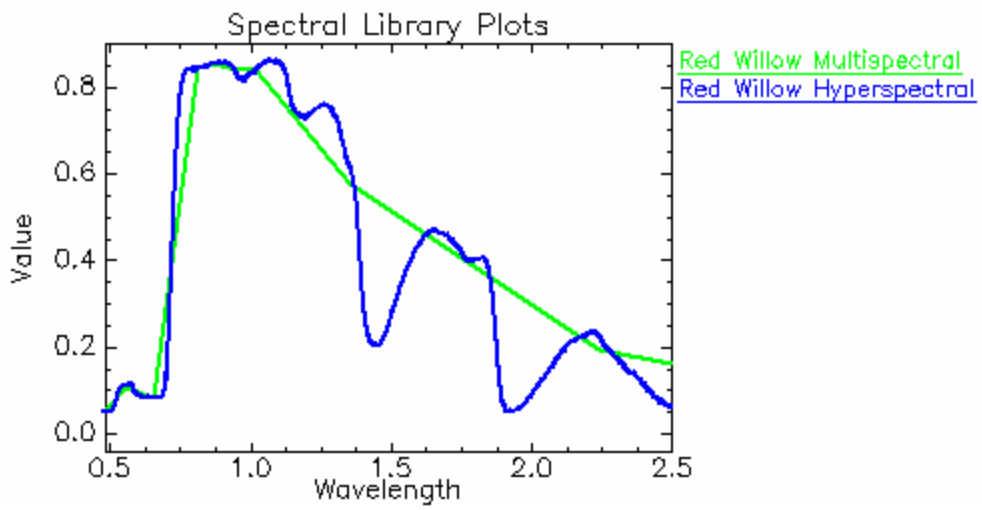

Figure 2: Hyper-spectral and multi-spectral vegetation signatures illustrating the "red edge".

Since multi-spectral sensors do not have the spectral resolution to discriminate between specific materials, they are typically used to differentiate between broad general classes of materials. The information content of hyper-spectral systems, however, allows them to be used to identify specific materials. Figure 3 illustrates how multi-spectral and hyper-spectral data are typically processed to identify general vegetation class and vegetation species, respectively.

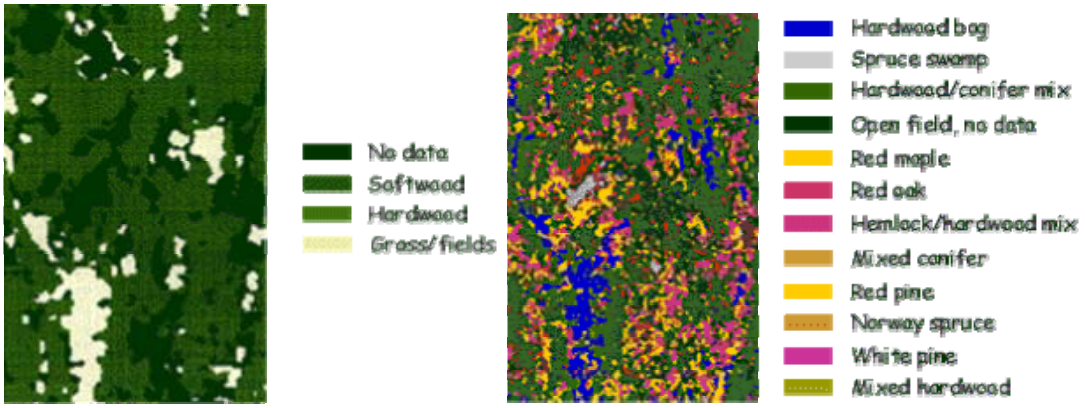

Figure 3: a) Multispectral classification versus b) hyperspectral discrimination [2].

2.3. Spatial aspect to imaging

The ability to identify spectral signatures of specific materials is highly dependent upon both the spectral and spatial resolution of the data. There are a limited number of photons reflected from the earth's surface. To provide adequate photons at the sensor, the ground sample distance (GSD) or ground pixel size must be balanced with the number of spectral bands. As detectors have become more sensitive, hyper-spectral instruments have become a reality. However they must integrate the signal over a larger spatial pixel to obtain sufficient signal in each of the narrow spectral bands. For the same reasons, single band panchromatic instruments have the best spatial resolution with multi-spectral instruments somewhere between these two extremes. Therefore, high spatial resolution imagery can significantly aid and complement the interpretation of high spectral resolution images, which necessarily suffer from poorer spatial discrimination.

Geology was one of the first disciplines to benefit from hyper-spectral remote sensing. Therefore, most theory is developed relative to earth material interactions. Increasingly, vegetation-based research is utilizing tools and techniques developed from geology. This type of successful identification from hyper-spectral sensors is highly dependent upon the GSD and the spatial extent of the materials of interest (e.g. how much area is covered predominantly by the material). Geologic formations tend to be large, thus providing good opportunities for the remote sensing geologic scientist.

Figure 4 illustrates the dimensions of a hyper-spectral data set. These data consist of two spatial and one spectral dimension that form what is typically referred to as a data cube. Whereas panchromatic images can be examined and interpreted directly by the analysts, the volume of information contained in the data cubes would be impossible to utilize without the extensive use of computers.

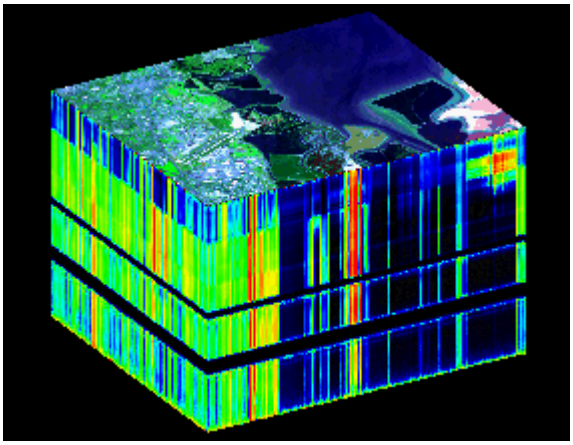

Figure 4: AVIRIS hyper-spectral data cube over Moffet Field, CA [3].

The data is further complicated because the instruments integrate signal over the full GSD. All satellite and airborne remotely sensed data are comprised of mixed pixels. The signal, or spectra, from each pixel in the image is a mixture of all the spectral signatures of all the materials within that area on the ground and in the optical path from the source to the detector. In a real scene, it is highly unlikely that even a single pixel will be comprised of anything pure. To address this issue there has been significant work in algorithm development in the area of unmixing. The details are beyond the scope of this paper but further study is available in [4] and [5]. New methods are being developed to "unmix" pixels without *a-priori* knowledge of the pure components known as spectral endmembers [6], [7]. However, if the amount of material in the pixel is sufficiently small, no algorithm will find that material.

3. Commercial satellite systems

Since the launch of the first civilian remote sensing satellite by the United States in 1972 carrying a multi-spectral sensor, the Earth Resources Technology Satellite 1 (ERTS-1), several spacecraft with increasing capabilities have been launched by the U.S. and by a number of other nations. The ERTS-1 was renamed Landsat-1 and all the subsequent remote sensing satellites in that family have been known as Landsat satellites. These are summarized in Table 2.

All the Landsat satellites and remote sensing satellite systems launched by other countries are placed in sun-synchronous near-polar orbits, see Figure 5.

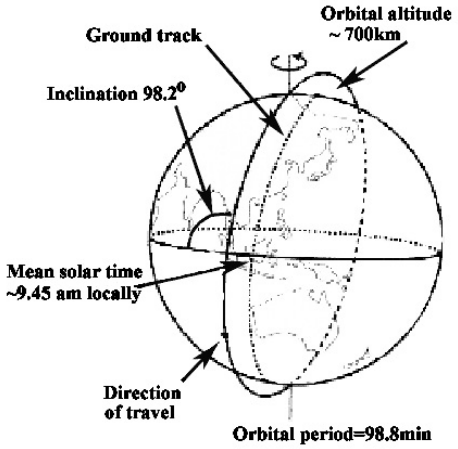

Figure 5: Orbit used by all Landsat-type remote sensing satellites.

3.1. Satellite imaging capabilities

The first three Landsat satellites carried Multi-Spectral Scanner (MSS) as the main imaging device with a Return Beam Vidicon (RBV) as a secondary instrument. On board tape recorders enabled satellites to acquire data globally. Landsat-4 and 5 carried a new sensor, the Thematic Mapper (TM) as well as the MSS. While these did not have onboard tape recorders, they relied on either direct transmission of data to ground receiving stations or the data was transmitted via the Tracking and Data Relay Satellite System (TDRSS). The TDRSS link on Landsat-5 is no longer operational. Landsat-7 has as its main sensor the Enhanced Thematic Mapper (ETM). The latter is similar to the previous two Landsat spacecraft except that it has a 15 m resolution panchromatic band that is co-registered with the multi-spectral data so that panchromatic and the multi-spectral images could be combined, enhancing the interpretation capability. Landsat-7 is capable of acquiring 250 day-lit images and transmitting them to a ground station via 150Mbps X-band link. The satellite also has a 375 GB solid-state memory, sufficient for 100 ETM scenes.

Table 2: Summary of the U.S. Landsat satellites [8], [9].

Satellite	Sensor	Band/ wavelength (μm)	Resolution (m)	Operational period
Landsats 1-2	RBV	1/0.48-0.57 2/0.58-0.68 3/0.70-0.83	80 80 80	Landsat 1 23/07/72-06/01/78 Landsat 2 22/01/75-05/02/82
	MSS	4/0.5-0.6 5/0.6-0.7 6/0.7-0.8 7/0.81-1	79 79 79 79	
Landsat-3	RBV	1/0.505-0.75	40	Landsat 3 05/03/78-31/03/83
	MSS	4/0.5-0.6 5/0.6-0.7 6/0.7-0.8 7/0.8-1.1 8/10.4-12.6	79 79 79 79 240	
Landsats 4-5	MSS	4/0.5-0.6 5/0.6-0.7 6/0.7-0.8 7/0.8-1.1	82 82 82 82	Landsat 4 16/07/82-07/87 Landsat 5 01/03/84- operational
	TM	1/0.45-0.52 2/0.52-0.60 3/0.63-0.69 4/0.76-0.90 5/1.55-1.75 6/10.4-12.5 7/2.08-2.35	30 30 30 30 30 120 30	
Landsat-7	ETM	1/0.45-0.52 2/0.52-0.60 3/0.63-0.69 4/0.76-0.90 5/1.55-1.75 6/10.4-12.5 7/2.08-2.35 Pan 0.50-0.90	30 30 30 30 30 150 30 15	15/04/99- operational

From Table 2, it can be seen that Landsat satellites have been improving in capabilities over a period of some 27 years. For example, Landsat-1 and 2 had RBV and MSS sensors with three and four spectral bands, respectively, and ground resolution of about 80 m. Landsat-3, on the other hand, had RBV and MSS sensors but with one and five bands respectively. The ground resolution of the former had improved to 40 m and the MSS had an extra channel (band 8) with a ground resolution of 240 m. Considerable improvement occurred in Landsat-4 and 5 with the introduction of the Thematic Mapper (TM) sensor. The ground resolution was improved to 30 m for bands 1 to 5 and 7 of the TM sensor and 120 m (compared to 240 m) for the thermal channel, band 6. However, the spectral bandwidth was considerably poor, as can be seen from Figure 6. In Landsat-7 a panchromatic band was introduced with a ground resolution of 15 m, sensitive to wavelengths between 0.5 μm and 0.9 μm, see Figure 6. Currently, the Landsat-7 ETM is not capable of producing full quality images because of the failure of the scan line corrector (SLC).

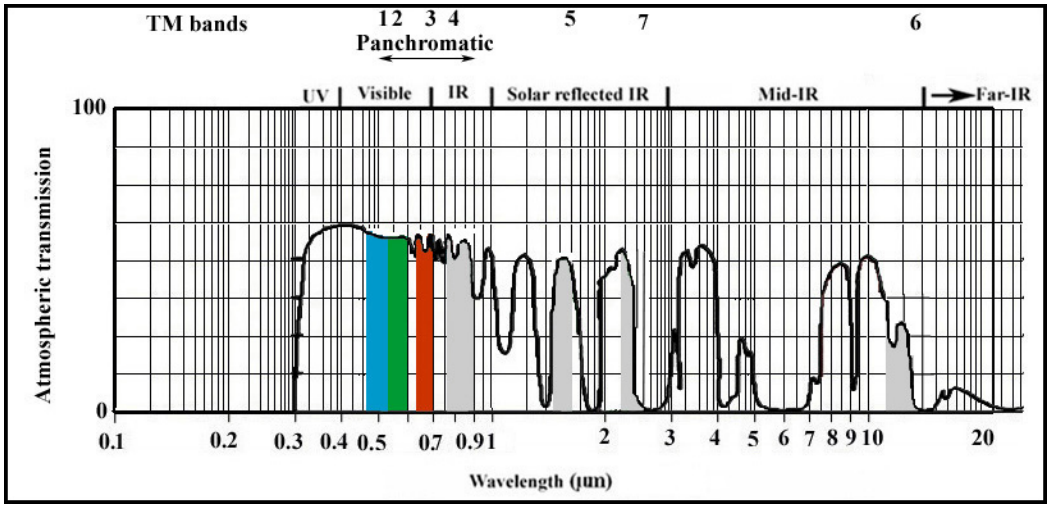

Figure 6: Bandwidth and regions of the electromagnetic spectrum in which the TM sensor is sensitive.

A substantive improvement came with the launch on 18 December 1999 of the U.S. ASTER (Advanced Spaceborne Thermal Emission and Reflection Radiometer) satellite. The characteristics of the sensors on board the ASTER are summarized in Table 3.

Table 3: Characteristics of ASTER satellite sensor **[10], [11**.

Sensor	Band number	Spectral range (μm)	Spatial resolution (m)	Quantization levels
VNIR	1 2 3N 3B	0.52-0.60 0.63-0.69 0.78-0.86 0.78-0.86	15	8 bits
SWIR	4 5 6 7 8 9	1.60-1.70 2.145-2.185 2.185-2.225 2.235-2.285 2.295-2.365 2.360-2.430	30	8 bits
TIR	10 11 12 13 14	8.125-8.475 8.475-8.825 8.925-9.275 10.25-10.95 10.95-1165	90	12 bits

The ASTER records imageries across 14 spectral bands with a swath width of 60 km and temporal resolution of 16 days. There are three sensors onboard, each collecting data in three different parts of the EM spectrum as shown in Figure 7. The visible and near infrared (VNIR) band has an improved spatial resolution of 15 m and band 3 has a backward looking capability inclined at 27.60° from nadir enabling stereo images. The thermal infrared (TIR) sensor has spatial resolution of 90 m much better than that obtained by Landsat-7. The short wavelength infrared (SWIR) band obtains data in 6 channels. The spectral resolution has improved to some extent. However, the duty cycle of the ASTER does not permit continuous data acquisition and therefore has to prioritize its data acquisition requests.

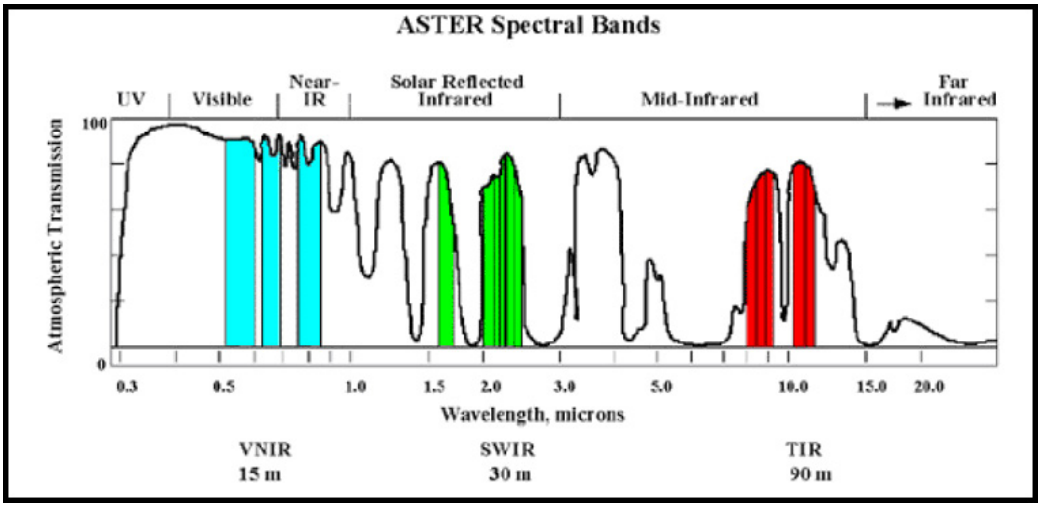

Figure 7: The spectral bands of ASTER sensors in relation to the EM spectrum.

While the spectral bands for the ASTER satellites have increased (14 bands compared with seven for Landsat-7), the spectral resolution is still limited. On 21 November 2000, the U.S. successfully launched its EO-1 satellite [12]. There are three basic earth imaging instruments on the EO-1 spacecraft, namely the Advanced Land Imager (ALI), the Hyperion (hyper-spectral imager), and Atmospheric Corrector (AC). The EO-1 ALI consists of a 15° Wide Field Telescope (WFT) and partially populated focal plane occupying 1/5th of the field-of-view, giving a ground swath width of 37 km. It provides seven multi-spectral bands from 0.43-2.35 µm with a spatial resolution of 30 m. The Hyperion is a high-resolution hyper-spectral imager capable of resolving 220 spectral bands (from 0.4 to 2.5 µm) with 30 m ground resolution. The instrument can image a 7.5 km by 100 km land area per image, and provide detailed spectral mapping across all 220 channels with high radiometric accuracy. The AC is an imaging spectrometer covering the spectral range from 0.9 to 1.6 µm and a spatial resolution of 250 m. This instrument is specifically designed to measure atmospheric information to assist in analysis of the Hyperion and ALI imagery. Some of the active spectral regions are summarized Figure 8.

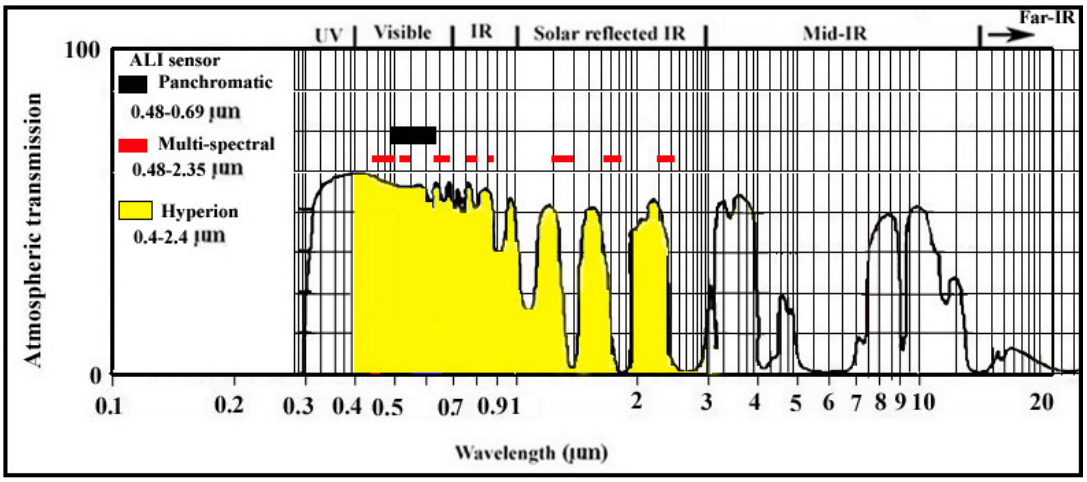

Figure 8: This shows some of the spectral bands of the EO-1 satellite sensors in relation to the EM spectrum.

A number of other satellites have been launched with panchromatic and multi-spectral sensors onboard. These are summarized in Table 4.

Table 4: This shows comparison between various commercial operational remote-sensing satellites.

SPOT-5		IRS-1C/D		Quickbird		Ikonos-2		Early Bird-1	
Band/ Res. (m)	**λ (μm)**	**Band/ Res. (m)**	**λ (μm)**	**Band/ Res. (m)**	**λ (μm)**	**Band/ Res. (m)**	**λ (μm)**	**Band/ Res. (m)**	**λ (μm)**
B1/10	0.50-0.59	B2/23	0.52-0.59	B1/2.44	0.45-0.52	B1/4	0.45-0.52	B1/15	0.49-0.600
B2/10	0.61-0.68	B3/23	0.62-0.68	B2/2.44	0.52-0.60	B2/4	0.52-0.60		
B3/10	0.78-0.89	B4/23	0.77-0.86	B3/2.44	0.63-0.69	B3/4	0.63-0.69	B2/15	0.615-0.670
B4/20	1.58-1.75			B4/2.44	0.76-0.90	B4/4	0.76-0.90	B3/15	0.790-0.875
Panchromatic									
Pan/2.5 or 5	0.48-0.71	Pan/5.8	0.50-0.75	Pan/0.61	0.445 -0.90	Pan/1	0.45-0.90	Pan/3	0.445-0.650

3.2. Future satellite systems

Whereas, the satellite community is discussing development of satellite systems with commercial opportunities, the authors know of no hyper-spectral systems currently under development. There has been discussion on the development of a European hyper-spectral sensor but its development has not moved passed the discussion phase.

4. Satellite imagery of uranium mines for safeguards

4.1. Past work

Although this paper is not intended as a thorough literature search, we will now discuss some examples of ongoing and related work.

There have been related studies on multi-spectral and hyper-spectral imagery for verification of uranium mines. They can roughly be divided into two categories: those that seek unique features of a uranium mine to possibly identify clandestine activities, and those that seek verification of State Declarations, such as movement of material. In [13], the authors investigated IKONOS multi-spectral and Hyperion hyper-spectral imagery to see if they could confirm operations, scheduling, and movement of materials. IKONOS was able to discriminate between different ore piles (i.e. group spectrally similar objects), while Hyperion detected possible high particle content in the tailings ponds. This information was used for change analysis.

In a similar paper [14], the objective was to confirm that operations reported by the States to the IAEA were consistent. Panchromatic images were used to compare the layouts, locations, consistency of scales with declared production levels, and operational status to those provided by the States. This paper identified discriminator stations as a unique feature of uranium mines, but pointed out that the station could not be differentiated from a refueling station from the images. For Ranger Mine in Australia, the size and extent of the sulfur and coarse ore stockpiles as well as traffic through the discriminator station are key features used to estimate the rate of uranium production. The multi-spectral and hyper-spectral images were used to look at the tailings in the ponds and determine if over a period of six months or longer these had changed, providing an indication of production rate of the mine.

Table 5: Proposed future satellite systems.

SYSTEM	# of Bands	Spatial Resolution	Country or Agency	Launch	Reference
Orbview-5		0.41 m panchromatic 1.64 m multi-spectral	U.S.	2007	www.orbimage.com
APEX (Earth Observing Instrument)/PRISM	>110 from .38 to 1 μm > 190 from .9 to 2.5 μm	2 – 5 m	European Space Agency (ESA)	Spring 2005	
Aries-1	96	10 m	Australia	2005	www.auspace.com.au/projects/aries.htm
CBERS3	4	20 m	Brazil and China	2008	www.cbers.inpe.br/en/programas/satelites4.htm
CBERS4	4	20 m	Brazil and China	2010	www.cbers.inpe.br/en/programas/satelites4.htm
EROS C	Multi-spectral	2.8 m	ImageSat International	2009	www.imagesatintl.com
NOAA-N/AVHRR	5	1.1 km			
TOPSAT	Pan Multi-spectral	2.5 m 5 m	UK	June 2005	http://www.sstd.rl.ac.uk/TOPSAT/technicalinfo.htm
ALOS (Advanced Land Observing Satellite) carrying AVNIR-2 (Advanced Visible and Near Infrared Radiometer type2)	4	10 m	Japan Aerospace Exploration Agency	Autumn 2005	

Previous studies involving the detection of uranium mines can be found in [15], where some spectral signatures of a uranium mine are presented, and [16], which seeks to identify uranium mine tailings. In the second paper, an airborne hyper-spectral sensor, Probe-1, took images over the Pronto mine tailings near Elliott Lake, Ontario, Canada, in 1999. The goal was to determine if uranium mine tailings could be distinguished from other types of mine tailings using unique mineral compound absorption features. However, no such features were discovered.

4.2. Proposed approach

In the case of uranium mining activities, the abundance of uranium in the ore can be 0.3% and significantly less in the tailings. Even under optimal circumstances, using currently deployed space systems to discern uranium will be challenging.

Therefore, approaches other than direct compound tracking will be needed. It may be necessary to consider an ensemble of observables that might provide confirmation of uranium mining operations. Reducing the overall process to individual steps and identifying what steps, if any, are unique to uranium mining is first required in this approach. Then determining the appropriateness of monitoring these particular steps using currently available multi or hyper-spectral satellite systems needs to be evaluated.

4.2.1. Site examined

Because of the diversity of the types of mining operations and possible sensitivities with characterizing some mining activities, we will use the Ranger Mine, a well-known site that has a large amount of open

literature information available, to illustrate the approach we are proposing. Located in a monsoonal part of Northern Australia, Ranger Mine is an open pit operation. The removal of rock from Pit #1 has been completed, and Pit #3 is currently being mined. Production has been over 4000 tons per year since 1996.

Figure 9 is a 1998 aerial photograph showing Pit #1 partly underwater. The pit in the foreground is the current ore body.

Figure 9: Aerial view of Ranger Mine and plant [17].

4.2.2. Mining/milling process

The mining and milling process at Ranger is composed of six steps: (1) mining, (2) crushing and grinding, (3) leaching, (4) solid-liquid separation, (5) solvent extraction, and (6) precipitation and drying [18]. In step 1, rock is extracted from the open pit mine and loaded into a truck. Each truckload of ore removed from the open pit mine is sorted using a radiometric discriminator. Based on this radiometric measurement, which is indicative of the percentage of uranium in the rock, the material is delivered to a waste pile, to a particular stockpile, or directly to the primary crusher. In step 2, the crushing and grinding process, the uranium ore is reduced to a fine size (80% passing 0.174 mm) in order to increase leaching efficiency. Aqueous solution is added to the ore to create slurry. In step 3, the uranium ore slurry is leached with sulfuric acid in a series of ten air-agitated pachuca tanks. Pyrolusite is added as an oxidant to increase leaching efficiency. In step 4, solid-liquid separation, the pregnant leach liquor, containing a vast majority of the uranium, is separated from the solid waste material or tailings. Solid-liquid separation is achieved through the use of counter current decantation thickener tanks, a clarifying thickener tank, and sand filters. In step 5, solvent extraction, uranium is first extracted from the aqueous pregnant leach liquor in a series of four mixer-settler units with an organic kerosene phase containing an amine which selectively complexes the uranium. The uranium-loaded organic solution is then passed to another series of four smaller mixer-settler units. The uranium is transferred to an aqueous phase through a change in pH achieved through the addition of ammonia. Finally, in step 6, precipitation and drying, ammonium diuranate is precipitated as yellowcake from the loaded strip solution by raising the pH with ammonia. The precipitate is centrifuged and calcined in a multi-hearth furnace.

Until 1996 tailings from the treatment plant were placed in an engineered dam, but they are now being deposited into the worked out Pit #1. No process or other contaminated water is released from the site.

Uranium mines and mills share many elements and processes with other types of mineral mines and mills, most notably copper [19]. Due to the close similarity between the Ranger mine and copper mining and milling operations, such as the Nchanga tailings leach plant in Zambia, distinguishing between the two might be problematic [20]. Therefore, this paper deals only with the comparison of uranium to copper mining and milling. Table 6 lists potential observables associated with each of the six steps at the Ranger mine and mill and also designates whether those observables are associated

with copper mining and milling. In the table, 'Exposed' refers to whether the observables are contained in covered or uncovered structures.

Table 6: Potential observables present at Ranger uranium mine and mill.

6a. Uranium mining

Target	Potential Observables	Size	Exposed	Present in Copper Mine/Mill
Open pit mine	Ore composition – Mg chlorite, quartz, sericite, 0-0.5% U minerals	437000 m^2	Yes	Yes
Ore stockpile	Ore composition – Mg chlorite, quartz, sericite, 0.2-0.5% U minerals		Yes	Yes
Gangue stockpile	Composition – Mg chlorite, quartz, sericite		Yes	Yes
Discriminator station	Discriminator building	5 m by 8 m	Yes	No (difficult to distinguish from other buildings)

6b. Uranium milling: crushing and grinding circuit

Target	Potential Observables	Size	Exposed	Present in Copper Mine/Mill
Primary crusher	Primary crusher building	700 m^2	Yes	Yes
Fine ore crushing station	Crushing station structure	1230 m^2	Yes	Yes
Coarse ore stockpile	Ore composition – Mg chlorite, quartz, sericite, 0.2% U minerals	7780 m^2	Yes	Yes

6c. Uranium milling: leaching

Target	Potential Observables	Size	Exposed	Present in Copper Mine/Mill
Neutral thickener	Surface water containing dissolved solids, flocculant	39 m diameter tank	Yes (occurs in open tank)	Yes
Leaching	1. Sulfuric acid, 2. oxidant (manganese dioxide, MnO_2)	Ten 7 m diameter pachucas	No (occurs in closed tanks)	1. Yes, 2. No
Sulfur stockpile	Elemental sulfur	9100 m^2	Yes	Yes
Sulfuric acid plant	Sulfuric acid, sulfur dioxide		Yes	Yes
Sulfuric acid tanks	Sulfuric acid	Three 15 m diameter tanks	No (sulfuric acid stored in closed tanks)	Yes
Pyrolusite stockpile	Manganese dioxide (MnO_2)		Unknown	No

6d. Uranium milling: solid-liquid separation

Target	Potential Observables	Size	Exposed	Present in Copper Mine/Mill
Thickeners (counter current decantation)	1. Pregnant leach liquor (PLL) (increased concentration of U), 2. tailings	Six 39 m diameter tanks	Yes	1. No, 2. Yes
Clarifying thickener	PLL	24 m diameter tank	Yes	No
Sand Filters	PLL	18 m diameter tank	Unknown	No

6e. Uranium milling: solvent extraction

Target	Potential Observables	Size	Exposed	Present in Copper Mine/Mill
Four extraction mixer-settler units	1. Kerosene, 2. trialkylamine (3.5%), 3. isodecanol (1.0%), 4. concentrated U	Four 420 m^2 units	No	1. Yes, 2. Yes, 3. Yes, 4. No
Four smaller strip mixer-settlers	1. Ammonia, 2. concentrated U	Four 110 m^2 units	No	1. Yes, 2. No
Ammonia storage tanks	Ammonia	Four 93 m^2 tanks	No	Yes

6f. Uranium milling: precipitation and drying

Target	Potential Observables	Size	Exposed	Present in Copper Mine/Mill
Precipitation tank	1. Ammonia, 2. yellowcake		No	1. Yes, 2. No
Ammonium Diuranate thickener	Yellowcake		No	No
Centrifuge	Yellowcake		No	No
Multi-hearth furnace (calciner)	Heat (800 degrees Celcius)		No	Yes
Yellowcake storage	Steel drums		No	No

Potential observables which are present in the Ranger uranium mining and milling operation, but not in copper mining and milling, include the discriminator station, pyrolusite (manganese dioxide) which is used as an oxidant in leaching, the pregnant uranium leach liquor produced in the sulfuric acid leaching process, the concentrated uranium strip solution generated from solvent extraction, and finally the yellowcake produced from the precipitation and drying steps.

4.2.3. Remote identification of observables

The remote identification of observables at Ranger which are characteristic of uranium mining and milling activity is complicated by the fact that these observables are of limited spatial size, may have indistinct spectral features, and may be hidden within covered buildings and structures. As an example, the discriminator station is approximately 5 m by 8 m in size and cannot be resolved in a Landsat or ASTER image as it will be contained in less than a single pixel at 80 m or even 15 m spatial resolution. The discriminator station could, however, be spatially resolved using a sensor with 1 m or better spatial resolution such as QuickBird. Further complicating the identification is the fact that the discriminator station does not have a unique spectrum. The spectrum is dependent on the building material used to construct the discriminator station. The oxidant pyrolusite is indicative of uranium leaching. However, the pyrolusite stockpile may be contained within a covered structure, preventing remote identification of this observable. The pregnant uranium leach liquor is visible in open tanks in the solid-liquid separation circuit, most notably in the six 39 m diameter counter current decantation tanks and the 24 m diameter clarifying thickener tank. Using a satellite system such as ASTER or Hyperion, the pregnant uranium leach liquor will be represented by only a handful of mixed pixels. The concentration of uranium in the pregnant leach liquor is also relatively low, on the order of 2 g/L U_3O_8, and, therefore, uranium may not be detectable using a satellite-based sensor with limited signal-to-noise. The concentration of uranium in the concentrated strip solution, at approximately 25-40 g/L U_3O_8, is considerably higher than in the pregnant leach liquor. However, solvent extraction occurs within covered structures, preventing direct spectral identification. Yellowcake, the final product of the uranium mining and milling process, is precipitated and dried in covered buildings, once again preventing direct spectral identification.

To address these limitations it may be necessary to utilize other sources of information to guide the hyper-spectral analysis. For example, high spatial resolution (< 1 m) panchromatic images could be used to identify relevant observables in the scene and these locations could be mapped back to the hyper-spectral images.

It should also be noted that there are large observables at Ranger that can be readily identified in satellite-based hyper-spectral images and are indicative of the chemical processes used at this

uranium mill. For example, the sulfur stockpile is spectrally distinct and encompasses multiple pixels in a 30 m resolution Hyperion image. The presence of the sulfur stockpile is indicative of the fact that sulfuric acid leaching is used at the Ranger mill, as the sulfur is converted to sulfuric acid at an on-site plant. While sulfuric acid is not a unique indicator of uranium milling activity as it is used in both uranium and copper leaching, the identification of the sulfur stockpile almost certainly rules out the possibility of gold or silver milling at the site as gold and silver are principally leached using cyanide.

5. Resource needs: analysts and tools

The individuals who compose an analysis group whose primary mission is the interpretation of *panchromatic* images is distinctly different from those individuals in remote sensing group whose goal is to interpret *spectral* remote images. The spectral remote imaging person will need to have a degree in remote sensing or comparable experience with knowledge of optics, detectors, atmospheric physics, chemistry, and physics. These people may be difficult to obtain, due to the limited pool of candidates. Currently, there are only a few remote sensing graduate programs, the most notable are in Tucson, Arizona, U.S., Seattle, Washington, U.S., and Toulouse, France.

Because there are currently no commercial algorithms or tools that specifically address the uranium mining/milling process, the remote sensing group must develop the expertise in-house. Since this application is not a commercially viable endeavor, successful identification will depend upon specialized algorithms developed for this application so a software developer will also be an integral part of the team.

6. Conclusions

Monitoring of uranium mining and milling processes using remote sensing systems could be a useful tool in the hands of international monitoring agencies to verify declarations and to look for undeclared activities. The current generation of multi-spectral and hype-spectral sensing platforms may provide some information related to open pit mining activities but no direct uranium information has been identified to date. Early research suggests the need to combine observables from various steps in the process as well as data obtained from different technologies (i.e. combining panchromatic images with multi-spectral information) to guide analysis and confirm activities. Observables will need to be present in large volumes due to the large spatial resolution of currently available sensors. This large spatial resolution may change with the next generation of sensor systems. However, when such systems would be built and deployed is currently unknown. The costs and resource demands of establishing a hyper-spectral satellite imagery group need to be weighed against what information could be gained with this technique versus information available from other sources, such as the existing panchromatic systems and in field inspections.

7. References

1. http://www.cameco.com/uranium_101/uranium_science/uranium/index.php
2. http://web.engr.oregonstate.edu/~vega/research/SSL_RS.ppt
3. http://aviris.jpl.nasa.gov/html/aviris.cube.html
4. Boardman, J.W., *Analysis, understanding and visualization of hyperspectral data as convex sets in n-space,* Proc. SPIE, vol. 2480, pp. 14 – 22.
5. Nielsen, A.A., *Spectral Mixture Analysis: Linear and Semi-parametric Full and Iterated Partial Unmixing in Multi- and Hyperspectral Data*, Journal of Mathematical Imaging and Vision 15, pp. 17-31, 2001.
6. Bowles, J., Palmadesso, P., Antoniades, J., Baumback, M., *Use of filter vectors in hyperspectral data analysis*, Proc. SPIE, vol. 2553, pp. 148 – 157.
7. Plaza, A., Martinez, P., Perez, R., Plaza, J., *Spatial/Spectral Endmember Extraction by Multidimensional Morphological Operations*, IEEE Transactions on Geosciences and Remote Sensing, vol. 40, no. 9, pp. 2025 – 2041, 2002.
8. http://landsat.usgs.gov/

9. http://www.mimas.ac.uk/spatial/satellite/landsat/characteristics.html
10. http://asterweb.jpl.nasa.gov/
11. http://edcdaac.usgs.gov/aster/asteroverview.asp#baseline
12. http://eo1.usgs.gov/sensors.php
13. Truong, Q.S. Bob, Borstad, G.A., Staenz, K., Neville, R., Leslie, R., Riggs, P., Bragin, V., *Multispectral and Hyperspectral Imagery for Safeguards and Verification of Remote Uranium Mines*.
14. Leslie, R., Riggs, P., Bragin, V., *Satellite Imagery for Safeguards Purposes : Utility of Panchromatic and Multispectral Imagery for Verification of Remote Uranium Mines*, Annual Meeting of the Institute of Nuclear Materials Management, Orlando, Florida, 23- 27 June 2002.
15. Neville, R.A., Staenz, K., Levesque, J., Nadeau, C., Truong, Q.S., Borstad, G.A., *Uranium Mine Detection using an Airborne Imaging Spectrometer*, presented at the Fifth International Airborne Remote Sensing Conference, San Francisco, CA, 17 – 20 September, 2001.
16. Levesque, J., Neville, R.A., Staenz, K., Truong, Q.S., *Preliminary Results on the Investigation of Hyperspectral Remote Sensing for the Identification of Uranium Mine Tailings*.
17. http://www.uic.com/au/emine.htm
18. *Ranger Uranium Mine*, Mining Magazine, pp. 392-403, November 1985.
19. Gupta, C.K., Mukherjee, T.K., *Hydrometallurgy in Extraction Processes*, Volume I and II, CRC Press, Boca Raton, FL, 1990.
20. Holmes, J.A., Stewart, L.N., Deuchar, A.D., Parker, J.D., *Design, Construction and Commissioning of the Nchanga Tailings Leach Plant*, 1976, in Extraction Metallurgy of Copper, Volume II, Hydrometallurgy and Electrowinning, Chapter 46, pp. 907-925.

GIS-based Systems to Support Open Source Information Analysis

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Abstract:

Open source information is widely used at the Agency in the State evaluation Process. The large amount of information to be considered calls for the use of powerful support tools for information retrieval and analysis. The GIS is suitable to deal with raster/vector maps and satellite images, and a Document Management System (DMS) is used to efficiently manage large sets of documents. Since many documents of interest concern nuclear facilities, i.e. entities with a geographic location, it appeared useful to explore the advantages of accessing a DMS by means of a GIS about Nuclear Fuel Cycle (NFC) facilities. The final product adds, to the powerfulness of the DMS, the possibility to query from maps, to display the results on maps and to facilitate the interpretation and synthesis of relevant information. In this paper we present the main features of the prototype system obtained by interfacing NUMAS, our geodatabase on NFC, with Verity, the DMS which manages our repository of documents on non proliferation.

Keywords: GIS, nuclear fuel cycle, additional protocol, satellite images, open source information.

1. Introduction

The verification activities carried out by the IAEA on country declarations make use, among others, of open source information, i.e. press, scientific journals, conference proceedings, Internet, etc. Satellite images are also important for the layout verification of declared sites as well as for supporting the evidence about hidden nuclear activities / undeclared sites.

The Nuclear Safeguards Unit of the JRC-Ispra started in 2000 to organise an Information and Analysis Centre (IAC) with the purpose of collecting information about Nuclear, Chemical and Biological Non Proliferation, Disarmament and Weapons of Mass Destruction, and producing studies on related subjects. Due to the large number of documents with similar information content it is frequently difficult to identify the original source. It is also difficult to consider such information sufficiently reliable to be used as they are provided. Therefore, the information collected is analysed by experts in the field who produce, on a regular base, internal documents on subjects of interest [1].

Being the collection and analysis of documents a heavy task, there is the clear need to use supporting tools. The commercial Verity Search 97 software [2] has been installed to classify, index and query the documents stored into IAC. Search 97 is a very powerful tool to manage large sets of documents.
Since many documents of interest are related to nuclear facilities, it appeared useful to explore the usefulness of accessing IAC's documents by means of NUMAS, a geodatabase on the nuclear fuel cycle facilities [3]. The addition of NUMAS to IAC allows adding, to the powerfulness of Search 97, the possibility to query from maps, to display the results on maps and to facilitate the identification of relevant documents. The aim of this paper is to describe the main features of the current version of NUMAS and

its interface with IAC through Search 97 to show how the identification of relevant documents could be facilitated by the use of a GIS interface.

Certainly, the effective retrieval and analysis of the smallest number of documents with the largest information content is a challenging research topic for language technologies specialists and, as such, it is faced by other scientists at the JRC [4].

2. The Verity system and its use with IAC

An important task of the JRC is to provide scientific and technical supports to the political and legislative bodies of the European Union (EU), in particular - in the context of the common EU policy - Non Proliferation issues, Disarmament and WMD are of growing importance to EU institutions. The availability of an Information and Analysis Centre, adding the scientific and technical perspective to the purely legislative one for the benefit of the policy makers at all levels, is without doubt very useful.
The tasks of the Information Centre were identified as follows [1]:

- Collecting and updating information concerning Treaties, Agreements, Regulations and the various Governmental Organisations (GOs) involved in the implementation. An additional set relates to Non Governmental Organisations (NGOs).
- Establishment of a competence centre on present and future material cycles (nuclear fuel cycle, chemical and biological industries, technologies, developments) and monitoring activities (NP, Disarmament and WMDs) worldwide.
- Gathering information mainly from open sources, thus allowing experts to produce studies on related subjects, such as country profiles.
- Evaluating, on the basis of the technological evolution, the improvements expected in the field of Non-Proliferation, Verification Regimes and Disarmament.

The data collected are organized using the Verity Search 97 system that allows handling large volumes of data in a variety of formats performing efficient indexing, searching, retrieving and displaying of textual information.

To adapt Search 97 to the IAC needs new modules have been developed [5]. Each document considered relevant by the expert is summarised and, together with metadata e.g. source, category, language, etc, is stored into the database of Sources of information.

Search 97 uses an advanced technology that allows the user to *search for concepts in documents*, rather than individual words or phrases, i.e. specific words and phrases in a document are considered as evidence of the presence of a concept. Keywords describing the concept are encapsulated in an entity called *"Topic Tree*" (TT).
Each concept is exploded into sub-concept and so on, leading to the construction of a tree-like structure. A set of simple and complex operators (logical, proximity, evidence, etc.) are used to create the topic tree. Therefore, a topic tree is a set of information describing a particular concept and structured as a predefined query that supports very efficient searches [6].
To query the collection of documents through the Search 97 interface (See Figure 1) the user enters from the keyboard a set of

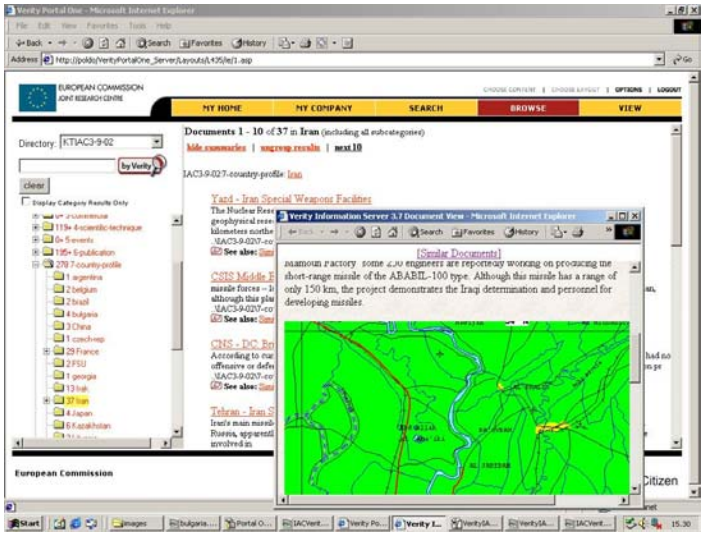

Figure1: Main interface of Verity Search 97

keywords by means of the query builder. Then Search 97 verifies, for each keyword, if there is a TT having that keyword as root; if yes the TT is executed as a complex query. The result is a list of documents with the associated score.

Importance weights can be assigned to sub-concepts / keywords which are used by Search 97 to calculate the score to assign to each document in the resulting list.
For instance, *s*uppose that the user is interested in retrieving documents about a given type of facility (fff) in operation, using a given process (ppp) and located in three countries xxx, yyy, and zzz. Suppose also that he/she wants to put the accent on the type of process more than on facility/countries; then the entered query could be e.g. [30](fff)and[*50*](ppp)and[*20*]((xxx)or(yyy)or(zzz)). Since Verity understands that the keyword (ppp) is more important than the others, it calculates the score and assign it to all retrieved documents, on the basis of the user-defined weights.

Topic trees are needed for all Nuclear Fuel Cycle (NFC) activities, processes, materials and equipments. Comparisons have been done on searching documents with and without a topic tree developed for the enrichment process [7]. Results have shown that without the topic tree a lot of documents related to "*enrichment"* in non nuclear field were found, which obviously were useless, whereas with the topic tree the search resulted in a fewer number of more specific documents.

It should be stressed that the search of documents in the IAC system is different from the search on the Web. In fact the former is carried out using Search 97 whereas for the latter different tools are used to extract from a text the relevant information [3]. The EMM, European Media Monitoring software, developed at the JRC, is aiming at efficiently searching for relevant information from newspapers. This tool has been adapted to the purposes of daily searching for information about non-proliferation on the press in the world [8]. Also the search of documents by means of EMM is based on a set of keywords specific to non proliferation issues.

3. The NUMAS system

The reasons for developing a new tool based on a Geographic Information System platform were threefold: 1) mapping nuclear fuel cycle activities (NFCA) in the world; 2) supporting the interpretation of satellite images of nuclear sites; 3) study the usefulness of retrieving IAC documents from maps.

1. Several databases on nuclear activities are available on the Web for consultation, but none of them offers the possibility to perform spatial queries and spatial analysis to, e.g. list all countries having a complete nuclear fuel cycle, or a specific part of it, show the spatial distribution of future plants or of decommissioned plants, etc. These Web sites have been very useful as open source information to feed NUMAS and to keep it up-to-dated.

2. The satellite image analysis, interpretation and reporting of nuclear activities is also very important. Here the main interest is not so much in the implementation of modules for image analysis, e.g. rectification, change detection, classification, which are well established, but rather to support image interpretation. How the different types of nuclear plants are seen from space? Which characteristic features allow recognising them? How can their status be identified?

3. NUMAS, as a mapping system for the NFC activities could also be conveniently used as a frond-end geographical interface to IAC, to facilitate the access to and the selection of documents relevant for subsequent analysis.

Each of the above capabilities are interdependent, in the sense that e.g. to keep the NFC mapping up-to-dated use is made of open source documents; to interpret a satellite image there is both the need to identify its location on a sufficiently detailed map as well as to use open source documents to know more about the site.

The main modules implemented in NUMAS are as follow:

- NFC Plant database;
- Tree-like structure of thematic layers;
- Satellite image analysis and interpretation;
- Query generator to access the DMS;
- Analysis of the characteristics of retrieved documents.

The NFC Plant database contains information about the different plants of the nuclear fuel cycle, from Mines & Milling to Waste Repository; it is not too different from the NFCIS database [9]. Main fields are Geographical coordinates, Country name, Site name, Owner, Operator, Plant type, Process, Material, Estimated annual production capacity, Maximum annual production capacity, Status, Start-up date, Closed down date.
The original content of NUMAS came from open source databases (IAEA-NFCIS [9], WANO [10], INSC [11]) and several other Web sites. Several differences were found in the geographical coordinates, (Latitude, Longitude) since generally the locations coincided with those of the closest town.
The objective of an on-going activity is to identify the exact geographical co-ordinates of sites through the analysis of open source information. The information about the exact location is important for various purposes, e.g. spatial analysis, location analysis, but also for more practical purposes, i.e. the purchase of high resolution satellite images.

With NUMAS the user can link any number of documents of any type (pdf, doc, xls, txt, tif, bmp, etc.) to any graphical object in any vector layer. This simple “local DMS” can be used to enhance the information associated with any element of vector maps. In particular raster/vector maps and satellite images cannot be treated efficiently by Search 97.

Moreover a tree-like structure of vector and/or raster maps can be associated with any graphical object in any vector map. Any object can be linked with a View. A View is a window in which several layers of information (raster and vector maps) can be organized within the same system of co-ordinates. Views can be linked together with simple clicks of the mouse.
This powerful mechanism gives the user to freedom to organize the information in the most convenient way. For instance, in a given site a View may be used to load the satellite image or the map of the area; on this view a vector layer can be drawn to highlight the different facilities in the site; then each facility can be linked to a second-level view containing a high resolution satellite image, documents/photos and/or to a third level view representing the layout of the facility, and so on. In this way any element on a map can be described at the desired level of detail.

The NUMAS modules dealing with satellite images analysis allow the user to perform basic operations on images such as rectification, change detection, and classification. An important requirement under implementation is a support to image interpretation. Based on the material produced in the context of an ongoing collaboration between JRC and EUSC (European Union Satellite Centre), an e-leaning system is under development containing information about the different sites and facilities of the NFC and how they are seen from space. Particularly, the identification from satellite images of elements typical of plant layout and operation are being collected to enrich the information content of the system. Several tests have been made on known European facilities and others on facilities known from

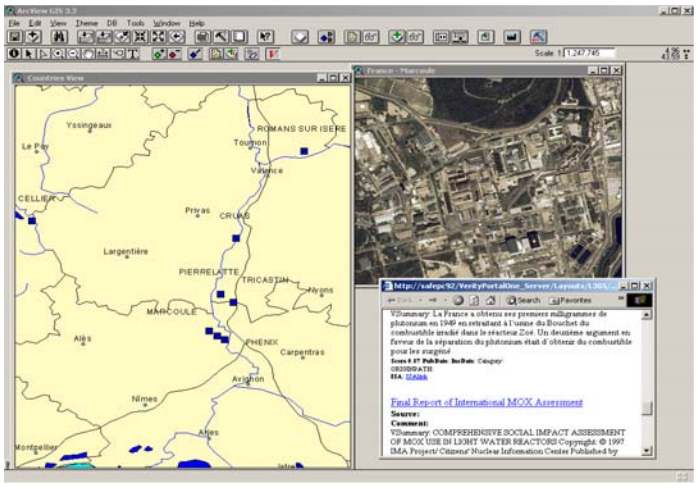

Figure 2: The NUMAS interface showing a View with a map and another View with a satellite image.

open source. Results from other researches, as for instance [12, 13] could be added to further enrich the information content. The software we use to manage this information is the EADS-Fleximage IMINT knowledge database [14]. This tool has interesting capabilities even though it presents some aspects that need further developments. If an image of a site has to be interpreted, it should be checked first of all if it refers to a known nuclear site, for which some information (satellite images, open source documents, in-house knowledge) is available. If it is unknown, it is interesting to see on a detailed area map what there is around it, e.g. infrastructures, other installations, populated areas, water, etc. and to search on the Web information about that location/activity.

4. The NUMAS - Verity interaction

A pilot study has been done to evaluate the usefulness of retrieving and analysing IAC documents through the NUMAS interface. Effectively, NUMAS can be used as a frond-end geographical interface to IAC to facilitate the access to and the selection of documents relevant for verification.
Figure 3 schematically shows the interactions between NUMAS and Search 97 carried out through the Query builder and the Documents' metadata module.

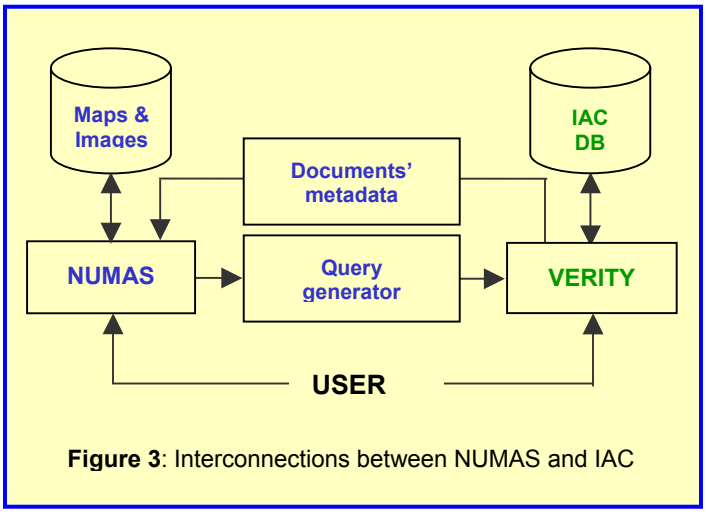

Figure 3: Interconnections between NUMAS and IAC

It is worth to stress the fact that a GIS, allowing searching for documents by clicking on a map, is very useful, provided that the documents of interest contain keywords referring to elements with a geographical location, e.g. country, site, facility, process.

With the use of a GIS interface the retrieval of documents is composed of two steps: 1) by a spatial query, i.e. a request whose result is represented on a new (dynamically generated) map; and 2) selection of the documents of interest. Spatial queries can be executed to find e.g. the spatial distribution of a given type of facility in a defined geographical area, type of process, material, status, or the NFC facilities in a given country. Then, as a second step, the Query Builder (See Figure 3) transforms the selection with the mouse of the graphical element(s) of interest into a query to Verity. The query is generated using as keywords the attributes describing the selected element. Then Verity interprets the query and, if Topic Trees with roots equal to the keywords exist they are triggered and used to select the documents of interest. In other words through the spatial query the user selects the sites for which the open source information, including satellite images, is of interest.

Consider again the example described in the previous section, concerning the facility type (fff) located in three countries (xxx, yyy, and zzz) and using the process (ppp). From NUMAS a spatial query can be performed (selection of keywords from menu) resulting in a map with the distribution of the fff-ppp facilities in operation in the world.
To get the documents, the user has different possibilities:
1) Click on the site hosting the type of facility of interest;
2) Click on a country;
3) Select the three countries xxx, yyy, zzz.
In the first case all documents related to the selected facility are retrieved; in the second case the retrieved documents concern all facility-process type in the selected country; in the last case as case 2 but applied to all selected countries.
The query automatically generated by the Query Builder module of NUMAS, is composed of the disjunction of combinations of keywords with the associated weights. Weights for each combination are

automatically determined on the basis of user-defined importance values associated to each keyword in the attribute database. As a general rule the importance increases as the keyword is related to a more detailed element, i.e. country has a lower weight than site; site has a lower weight than facility, and so on. Therefore, the manually constructed query, typical of Search 97, is substituted by the automatically generated query of NUMAS. The result of the query is the list of retrieved documents sorted by decreasing value of importance with the associated official source.

Moreover, countries in which the *sources of retrieved documents* are located are automatically highlighted on the world map, shown for instance in Figure 4 about enrichment facilities in operation, thus giving immediately to the user an idea about the countries interested in the searched topic. Then, clicking on a country map, the list of documents issued by institutes in that country is displayed, as shown in Figure 5. Other information could be extracted from the documents and displayed on a map as for instance the name of nuclear sites.
Filters on e.g. score, date of publication, documents' category can be applied to restrict the resulting list.

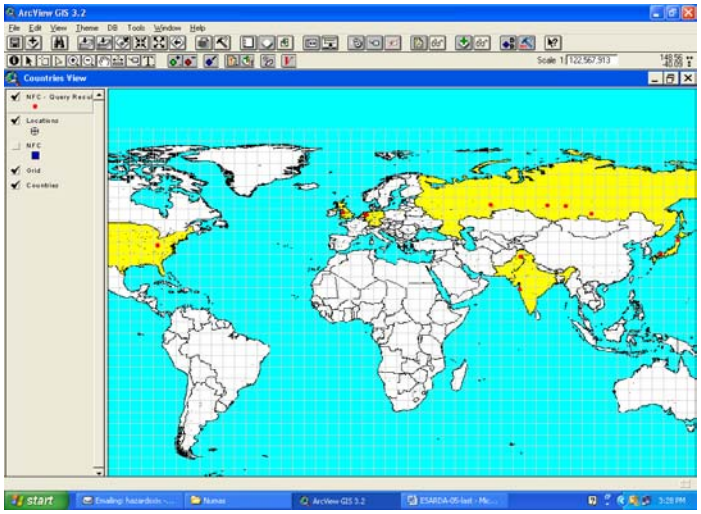

Figure 4: Map of enrichment facilities in the world based on the diffusion process.

Generally speaking, it can be said that the interface of a GIS with a DMS offers some advantages, e.g.:

— the spatial query works as a filter, allowing the user to better focus the search;
— the list of documents can be retrieved clicking on a graphical object or after selecting a group of elements;
— statistics, about e.g. the distribution of the number of retrieved documents vs. country, year of publication, language, can be calculated;
— some characteristics of the retrieved list of documents can be represented on a map, thus facilitating both the fine tuning of the search and the interpretation of results.

The last of the above points is still a research issue.

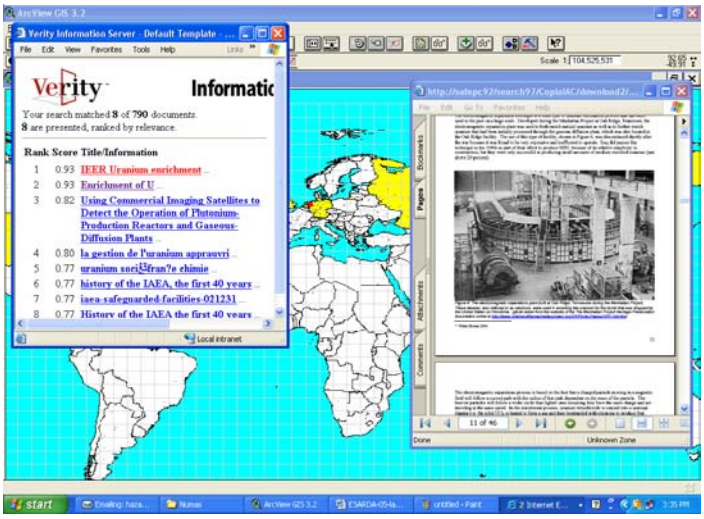

Figure 5: List of documents about enrichment facilities issued in the selected country.

The search for open source documents also helps in keeping the geodatabase up-to-dated. In fact, any time a new document is considered of interest it is stored into IAC; if it refers to a known site, it can be accessed via the geographical map; if it refers to a new site, the site itself is put on the country map, so that the document can be retrieved just clicking on the site location. Therefore, it can be conceived that when we receive a request for information on a given site, we are able to access the summary documentation, elaborated according to the user needs, as well as the original documents.
It is also possible to search for documents from a map without performing a specific spatial query. This is particularly useful when documents about a certain country / site are of interest.

5. Conclusions and further developments

In this paper we have briefly presented the main results achieved so far on an going project aiming at developing software tools for supporting the open source information retrieval and analysis. Some of the tools needed are based on a GIS platform and on a powerful DMS. A very important aspect concerns the application of language technology techniques to the retrieved documents. The combined use of various sources of information (open source documents, satellite images, in-house information, etc.) within the same software environment may produce values added information and/or triggers more focused searches.

6. Acknowledgements

The authors whish to thank Mr L.V. Bril for all fruitful discussions and Mr J. Goncalves for the support given to this work.

7. References

[1] LV. Bril, M. Cuypers, U. Meloni, F. Mousty, A. Poucet, European Non Proliferation Information Management and Analysis Center, Institute for Systems, Informatics and Safety (ISIS), JRC Ispra, Italy, May 2001.
[2] http://www.verity.com
[3] S. Contini, F. Bellezza, The NUMAS geodatabase, User manual, EC-JRC, In preparation.
[4] C. Ignat et al, A Tool for the Quick and Efficient Exploration of Large Documents Collections, this Seminar.
[5] A. Ussorio, Development of an Information and Analysis Centre, Non Proliferation and Nuclear Safeguards, Joint Research Centre, Ispra, January 2004.
[6] L.Costantini, J. Hill, Information Collection Strategies to Support Strengthened Safeguards, Symposium on International Safeguards, Verification and Nuclear Material Security, Internal Atomic Energy Agency, Vienna, October – November 2001.
[7] I. Maschio, Candidate countries – Country profile, Universita' Bocconi - JRC Ispra, Internal document, December 2003.
[8] http://emm.jrc.cec.eu.int
[9] http://www.iaea.org/DataCenter/datasystems.html
[10] http://www.wano.org.uk
[11] http://www.insc.anl.gov
[12] B. Jasany, P. Hoffman, Classifying Nuclear Power Plant Facilities Using Already Developed "Key", this Seminar.
[13] I. Niemeyer, Use of Thermal Infrared Image Data for Nuclear Safeguards Purposes, this Seminar
[14] http://www.fleximage.fr

Session 11

Measurement Techniques for Spent (and fresh) Fuel

Determining the nuclear-material content of a mixture of damaged spent fuel within sealed containers at the Paks NPP

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Abstract:

Due to an incident at the Paks NPP in 2003, 30 fuel assemblies kept in a "cleaning tank" became damaged. Many fuel rods broke and different types of pellets fell out from the rods and got mixed with each other and with broken construction elements (e.g. cladding). The damaged fuel will be repackaged into closed canisters and kept in the spent-fuel pond until further action. Declaring the nuclear-material content of the individual closed canisters is a difficult task, because the canisters will contain a mixture of spent-fuel pieces of different initial enrichment, burn-up and irradiation history distributed in an irregular geometry. No measurement technique has yet been published concerning the quantitative assay of nuclear material in such physical form.
The purpose of this presentation is to describe the passive non-destructive method which will be used for determining the nuclear-material content of the closed canisters at the Paks NPP. The method is based on the combination of gamma-spectrometric and neutron-counting measurements. These measurements will provide the inventory to be declared. Specific equipment will be built and a novel methodology, described in this presentation, will be applied.
The measurements to be carried out could serve as a precedent for inventory-taking of nuclear material in a mixture of different types of nuclear and non-nuclear material distributed in an irregular, unknown geometry.

Keywords: spent fuel; nuclear-material content; NDA

1. Introduction

On April 10, 2003 an incident occurred in unit 2 of Paks NPP, during which 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 will be repackaged into closed containers, which will be kept in the spent-fuel pond until further action [5].

The purpose of this paper is to present the nuclear measurement techniques and procedures which are planned to be deployed in determining the nuclear-material content of the closed containers. The measurements will enable us to prepare the inventory of the nuclear material, which the NPP will report to the safeguards authorities. In the present document we use the terms "nuclear material" and "fissionable material" as synonyms to denote the uranium and plutonium in the spent reactor fuel. Other definitions and other types of "nuclear" or "fissionable" material (e.g. thorium) will not be considered in this work.

The damaged-fuel containers will be kept 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. In addition, the inventory will include an estimate of the amount of nuclear material dissolved in the water

or accumulated on the ion-exchange resin and as contamination on the surfaces of various objects in the spent-fuel pond, the loading pit and the cleaning tank.

The Paks NPP and the Institute of Isotopes (IKI[1]) signed a contract according to which IKI will determine the nuclear-material content of the closed containers and in this way assist in the preparation of the nuclear-material inventory of the damaged fuel. The completion of this task is divided into three phases.

In the first phase, which ended in September 2004, IKI created a draft document about the concepts of inventory taking of the damaged fuel and determining the nuclear-material content of the individual containers.

In the second phase, which ends in November, 2005, IKI will build the equipment necessary for the measurements. Furthermore, in this phase IKI will perform calibration and testing of the equipment using regular WWER-440 fuel assemblies. The equipment will contain neutron counting and gamma spectrometric devices and its construction will be similar to the "Enhanced Fork Detector" [6], [7], [8] and, to some extent, to the "SMOPY" device [9].

The third, last phase will commence as soon as the damaged fuel is loaded into the containers. During the recovery process the total mass of the material filled into each container will be documented by the company performing the loading of the containers. Together with the visual surveillance of the loading process, this will enable a rough estimation of the nuclear-material content of the containers and the preparation of a "preliminary inventory". This preliminary inventory will be verified or corrected by independent nuclear measurements carried out by IKI. In this way the amounts of ^{235}U, total U and Pu will be determined, which the NPP will declare to the safeguards authorities.

In section 2 of this paper the present status of the damaged fuel will be described and some details of the recovery process relevant to the inventory taking will be given. Section 3 contains the measurement principles of the nuclear measurement techniques to be used. In section 4 the procedure which will be used for verifying those containers that are supposed to be fissionable-material free will be described (i.e. their fissionable-material content is less than the smallest detectable amount). In section 5 the nuclear measurement techniques and the methodology which will be applied for determining the nuclear-material content of the damaged-fuel containers will be presented. Section 6 contains an overview of the measurements which are necessary in order to calibrate and test the equipment which will be used in inventory taking. A description of the mechanical construction of the device to be built for the measurements is given in section 7.

2. Status of the damaged fuel and details of the recovery process relevant to inventory taking

In the incident of April 10, 2003 at Paks NPP all the 30 fuel assemblies in the cleaning tank were damaged to some extent. Many 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] (see Fig. 1). 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.

There will be three types of containers used for the remnants of the damaged assemblies [13]. Two types are planned 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.

[1] The abbreviation "IKI" stands for "Izotópkutató Intézet", the Hungarian name of the Institute of Isotopes

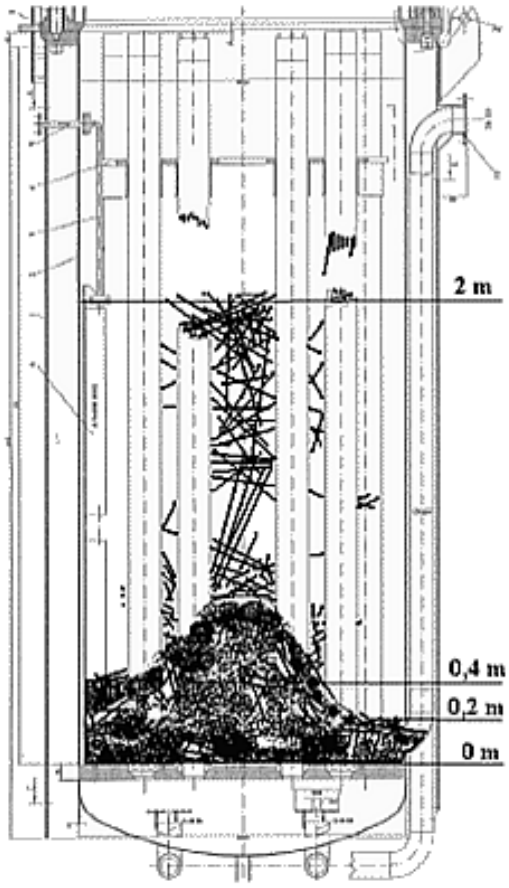


Fig. 1. Present status of the damaged fuel in the cleaning tank [10], [11], [12].

The upper bound for the mass of nuclear material within an individual container can be calculated from the total mass of the fuel loaded into the container, since the percentage of nuclear material within the damaged fuel can be estimated from the depletion calculations performed for the damaged assemblies. If one relies merely on the weight measurements one cannot be sure that there is any nuclear material within the container. So based merely on weight measurements, the lower bound for the mass of nuclear material in a container is 0 g. Therefore, if one adopts the mean value of the lower and the upper bound as the mass of nuclear material within the container, the error of such an estimate would be in both directions the half of the upper bound, i.e. $\pm$100%. Note however, that the weight measurements will be supplemented by visual information as well and it is likely to be noticed if no nuclear material is put into a container, so the above estimate for the lower bound might be too conservative. Nevertheless, the reliable verification of the contents of the containers is only possible by using nuclear measurement techniques, although this does not automatically imply that the nuclear measurements will be more precise than the weight measurements.

A few containers will be loaded with the assembly parts made of non-nuclear material, that is, with the heads and tails of the assemblies. These containers will be verified for the absence of nuclear material and then they will be removed from the spent-fuel pond as soon as possible, because of tight placement problems in the pond.

It should be emphasized that the nuclear material inventory will have to include an estimate concerning the amount of fissionable material dissolved into the pond water or accumulated on the ion-exchange resin or as contamination on the surfaces of various objects in the spent fuel pond, the loading pit and the cleaning tank. In the present paper, however, we will only concentrate on determining the fissionable-material content of the closed containers.

3. Principles of the applicable nuclear measurement techniques

The procedure for determining the nuclear-material content of the closed containers is based on a combination of neutron-counting and gamma-spectrometric techniques. Most of the gamma-photons get absorbed within the fuel so they carry information merely about the outer regions of the fuel. In order to see deeper into the fuel volume, we will also count the neutrons emitted by the spent fuel. Namely, the absorption of neutrons within the fuel is much less than that of the gamma-photons, therefore using neutron counting one can gather information about the inside of the observed fuel volume. The drawback of neutron counting is that the contribution of the individual isotopes to the gross neutron count can only be deduced if additional information (e.g. burn-up and cooling time) and depletion calculations are available.

The basic idea of the measurements is to determine the mass of each fissionable isotope within a container, m, as the product of the concentration of that fissionable isotope, ρ, and the total mass of nuclear material initially present in the observed fuel volume, $m_{ini}(U_{total})$,:

$$m = \rho m_{ini}(U_{total}) \quad (1)$$

We will use neutron counting to get information on the initial mass of uranium, characteristic for the observed volume of the fuel, and gamma spectroscopy to collect information on the effective burn-up. The burn-up will then be used to determine the concentration of the individual fissionable isotopes, through semi-empirical correlations supported by depletion calculations (assuming the average concentration and burn-up to be the same in the outer and inner regions of the fuel). Combining these pieces of information, the mass of nuclear material within the observed volume will be calculated (section 5.3). The total mass of nuclear material within the container will be determined by scanning along the full length of the container, from at least 3 of its sides.

3.1. Gamma spectroscopy

In the case of spent fuel, the information about the fissionable-material (i.e. U and Pu) content of the fuel cannot directly be extracted from the gamma lines of the relevant U and Pu isotopes, because these lines are masked by the much stronger gamma peaks of the fission products. It can be seen in Fig. 2 that no peaks of the U and Pu isotopes are visible. On the other hand, some fission products can clearly be distinguished. Therefore, one should apply indirect methods based on measuring the activities of appropriately chosen fission products. The relevant properties of the selected fission products are given in Table 1.

Fission product	Fission yield, %			Neutron capture product	Half life, days	Gamma-energies, keV (Emission probability, %)
	^{235}U	^{239}Pu	^{241}Pu			
^{144}Ce-Pr	5.48	3.74	4.39		284.9	696.5 (1.34); 1489.2 (0.3); 2185.6 (0.7)
^{106}Ru-Rh	0.4	4.31	6.18		372.6	511.9 (20.7); 621.9 (9.8); 1050.5 (1.6)
^{133}Cs	6.70	7.02	6.54	^{134}Cs	752.4	563.26 (8.38); 569.29 (15.43); 604.7 (97.6); 795.8 (85.5); 801.84 (8.73); 1038.6 (1.1); 1167.9 (1.8); 1365.1 (3.0)
^{153}Eu	0.161	0.365	0.555	^{154}Eu	3214	723.4 (19.7); 873.2 (11.4); 1004.8(17.4);1274.4 (35.5)
^{137}Cs-Ba	5.88	6.27	6.47		11008.6	661.6 (85.2)

Table 1. Relevant data of chosen fission products.

The gamma-spectroscopic information will be used to calculate the "effective" burn-up of the measured volume of the damaged fuel. The burn-up is calculated from the activity ratios ^{134}Cs/^{137}Cs and/or ^{154}Eu/^{137}Cs (see section 5.1). The effective burn-up will be used to obtain the average concentration of the fission products in the measured volume (section 5.2). The gamma-spectrometric information can also be used for verifying the cooling time of the fuel.

For the gamma-spectroscopic measurements of the spent fuel we considered the use of high purity Germanium (HPGe) and Cadmium-Zinc-Telluride (CZT) detectors. A gamma spectrum of a WWER-440 fuel assembly taken with a HPGe detector at Paks NPP is shown in Fig. 2.

In order to calculate the various activity ratios of the fission products from the gamma spectrum one should first determine the relative efficiency curve of the detector (intrinsic efficiency calibration method, when the relative efficiency curve of the detector and the activity ratio are determined from the same spectrum). In our case the gamma lines of ^{134}Cs, present in the spectrum of the spent fuel from 475 keV to 1365 keV, provide the means for establishing the relative efficiency curve.

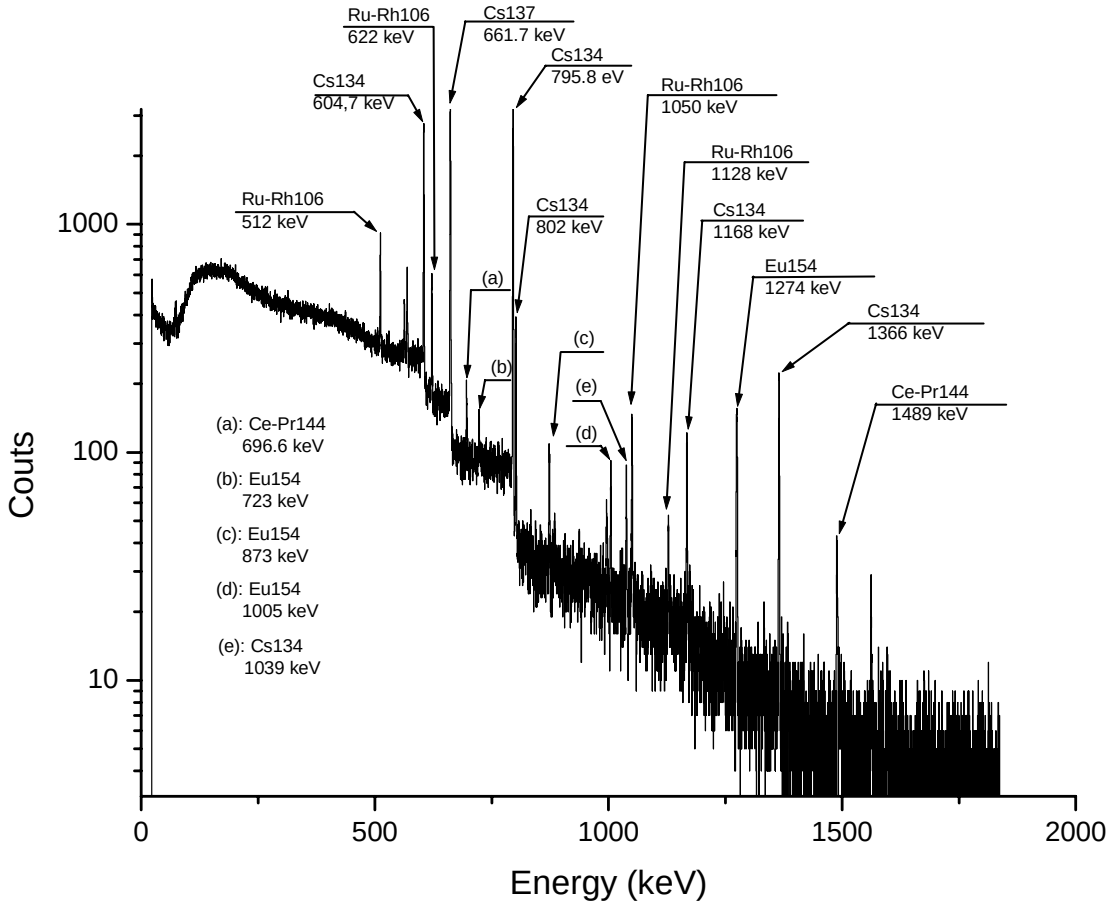


Fig. 2. Spectrum of a WWER-440 assembly of 3,6 % initial enrichment taken with a HPGe detector.

The fillings of the damaged-fuel containers will presumably be quite inhomogeneous along the length of the containers. Therefore one should scan the containers along their full length and measure the intensities of the gamma-lines of the fission products at as many points as possible. The more accurate and detailed the intensity profile of the main Cs-isotope gamma lines is, the higher the precision of the mass measurement will be. Obviously, the intensity profile of a container loaded with damaged fuel will be much more complex and very different from that of a regular fuel assembly.

3.2. Neutron-counting measurements

For the neutron-counting measurements, the use of two fission chambers and two bubble detectors is planned. They are practically insensitive to gamma photons and their output is proportional to the gross neutron yield. As indicated above, neutron measurements will have a role in gathering information on the initial mass of uranium, characteristic of the total observed volume of the fuel.

In our case the cooling time will be about 3 years (in the beginning of 2006). The absolute neutron emission rates for this type of fuel are given in Table 2 and Fig. 3, based on theoretical calculations for WWER-440 assemblies for a few given burn-up values [15].

Burn-up, GWd/tU	Neutron emission rate, n/cm^{3}s
10	1.3498×10^1
20	2.0732×10^2
30	1.3046×10^3
45	7.6748×10^4

Table 2. Absolute neutron emission rates in spent WWER-440 fuel of 3.6% initial enrichment, after 3 years cooling time, as a function of burn-up [15].

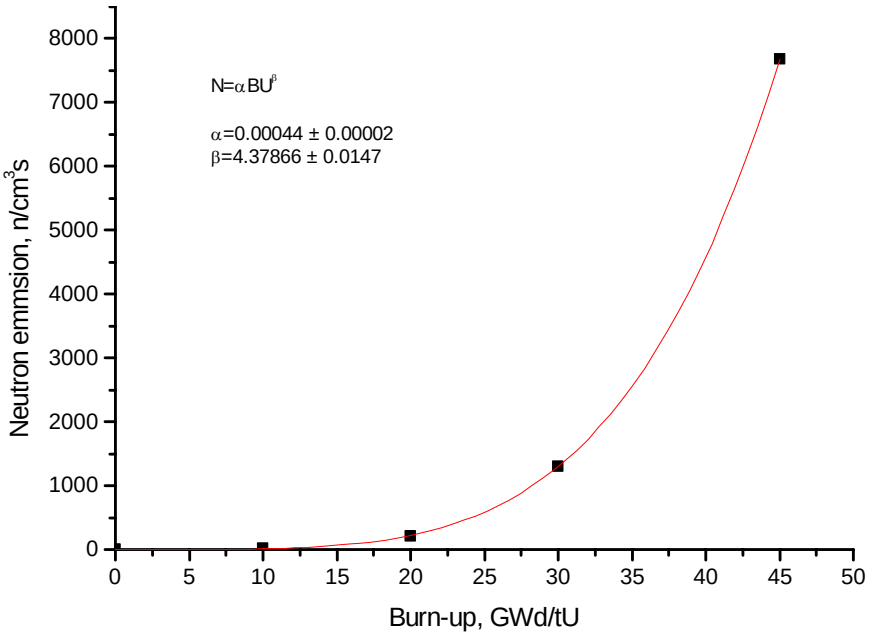


Fig. 3. Absolute neutron emission rates in spent WWER-440 fuel of 3.6% initial enrichment, after 3 years cooling time, as a function of burn-up [15].

The neutron measurements will be performed using fission chambers and bubble detectors simultaneously. Fission chambers are standard instruments for safeguards verification of spent fuel assemblies and plenty of practical information is available about their use. E.g. they are used in the FORK [1], [7], [8] and SMOPY [9] devices (see also references therein) for neutron measurements. Therefore, we will use fission chambers in our measurements as reliable and tested reference instruments. On the other hand, by bubble detectors, which are sensitive mainly to fast neutrons, a better spatial resolution can be achieved when scanning the neutron profile of an assembly or damaged-fuel container (Fig. 4). The better the spatial resolution is, the more accurate the calculated mass of the nuclear material will be. Therefore we will also use bubble detectors in our measurements.

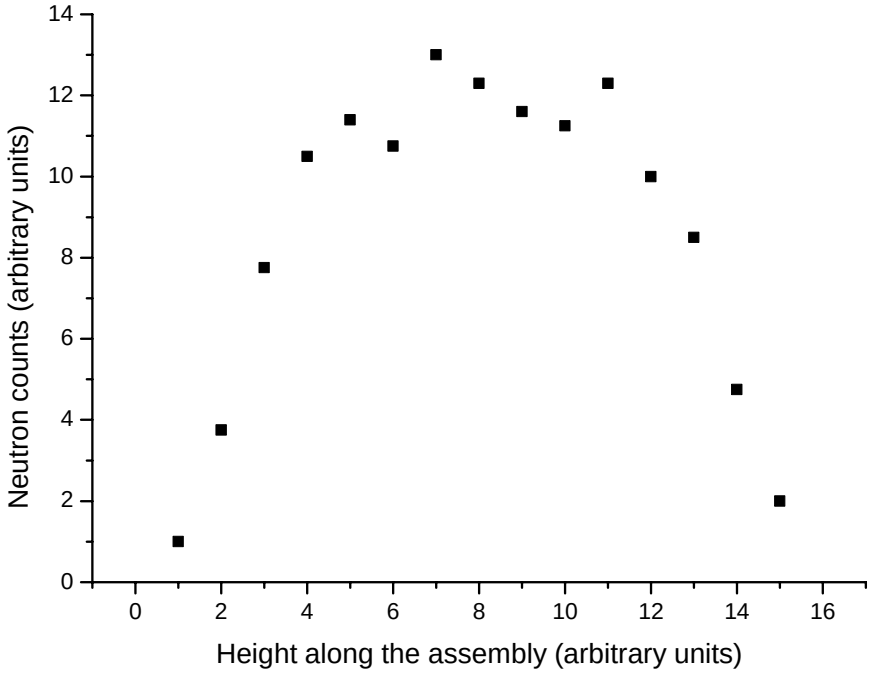


Fig. 4. The distribution of neutrons emitted from a WWER-440 assembly at Paks NPP, measured by a CR-39 track detector along the length of the assembly [16]. Track detectors are, like bubble detectors, also sensitive to fast neutrons, so the neutron profile obtained by them is similar to the one which can be obtained by bubble-detectors.

4. 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 verification will preferably be performed immediately after the containers are moved to the spent fuel pond from the loading pit. The design of the equipment used for verification will be similar to the “Enhanced Fork Detector” [6], [7], [8] and it will contain fission chambers and bubble detectors for neutron counting and a CZT detector for taking medium-resolution gamma spectra.

The containers will be scanned along their full length. The intensity of the gamma peaks of ^{137}Cs and the neutron count rate 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 their deviation from the background is larger than the (measurement time dependent) statistical error of the applied detector. The statistical error of the detectors is the most important factor influencing the smallest detectable value. In order to determine the smallest detectable value, the equipment will have to be calibrated using regular fuel assemblies of known nuclear-material content and at least one measurement will have to be performed on the containers filled with the damaged fuel. It should be noted that the smallest detectable value will also depend on the distribution of nuclear material in the container, i.e., on whether the nuclear material is concentrated in a small volume or distributed evenly across the full length of the container. Nuclear material concentrated in a small volume of the container, i.e. a bunch of pellets stuck in the tail of an assembly, might probably be detected, but a single lost pellet will most likely be undetectable.

5. Determining the nuclear-material content of the damaged-fuel containers

The composition and burn-up of the nuclear fuel before the incident in each assembly is known from depletion calculations. In the incident, however, nuclear material with different properties got mixed with one another and cannot be separated. In addition, its distribution within the containers will be quite irregular. Because of this, the damaged-fuel containers will be scanned along their full length, from at least 3 sides, and the measured gamma-peak areas and neutron counts will be averaged over all the observed sides. In this way the data will be collected at several measurement positions along the containers.

5.1. Determining the burn-up

The burn-up will be determined from the gamma-spectroscopic data. It will be calculated from the activity ratio ^{134}Cs/^{137}Cs and/or ^{154}Eu/^{137}Cs. The method relies on the facts that ^{137}Cs is produced immediately during fission, while ^{134}Cs is produced by neutron capture, from ^{133}Cs created by fission. Therefore the activity of ^{137}Cs is, in a first approximation, directly proportional to the neutron fluence (integral neutron flux), i.e. to the burn-up, and the activity of ^{134}Cs is proportional to the square of the neutron fluence (i.e. square of the burn-up). The activity ratio ^{134}Cs/^{137}Cs is thus proportional to the burn-up. The similar is true for the activity ratio ^{154}Eu/^{137}Cs. Note that for regular, undamaged assemblies the activity of ^{137}Cs would be sufficient in itself for determining the burn-up. Using the above mentioned activity ratios, however, many systematic errors can be avoided, and thus the accuracy of the measurement can be improved.

The burn-up is calculated form the formula

$$BU = K \frac{A(Cs134)}{A(Cs137)} \exp[(\lambda_{Cs134} - \lambda_{Cs137})t] \ , \qquad (2)$$

where t is the cooling time of the fuel, the λ-s are the corresponding decay constants and K is a calibration constant of the measurement device which will be determined using regular fuel assemblies. The activity ratio A(C134)/A(Cs137) is determined using the intrinsic calibration method from the gamma spectrum of the fuel. A similar equation can be written for the activity ratio A(Eu154)/A(Cs137) also, and it will be used for verifying the result obtained by formula (2).

It should be emphasized, that for the fuel mixed within the containers one can measure merely an ***effective*** burn-up, since this mixture contains spent fuel elements of various burn-up, which cannot be separated. Therefore, the value of the quantity calculated from formula (2) will not correspond to any particular burn-up of any of the spent-fuel pieces in the containers. Nevertheless, the effective burn-up obtained in this way can be used in determining the concentration of fissionable material within the containers, as described in section 5.2.

5.2. Determining the concentration of the fissionable material

By “concentration” of ^{235}U, ^{238}U and total Pu here we mean the ratio of the mass of ^{235}U, ^{238}U and total Pu in the spent fuel to the initial mass of total uranium. We will determine the concentration of fissionable material in the damaged-fuel containers using experimentally verified correlations between the burn-up and the amounts of ^{235}U, ^{238}U and total Pu remaining in the spent fuel. The correlations were obtained by depletion calculation codes for WWER-440 assemblies and are given in reference [15], for various initial enrichments. The results of the calculations given in reference [15] for assemblies of 3.6% initial enrichment, relevant to our measurements, are presented graphically in Fig. 5, Fig. 6 and Fig. 7, together with the data points obtained from the depletion calculations preformed using the data of reference assemblies and of the damaged assemblies.

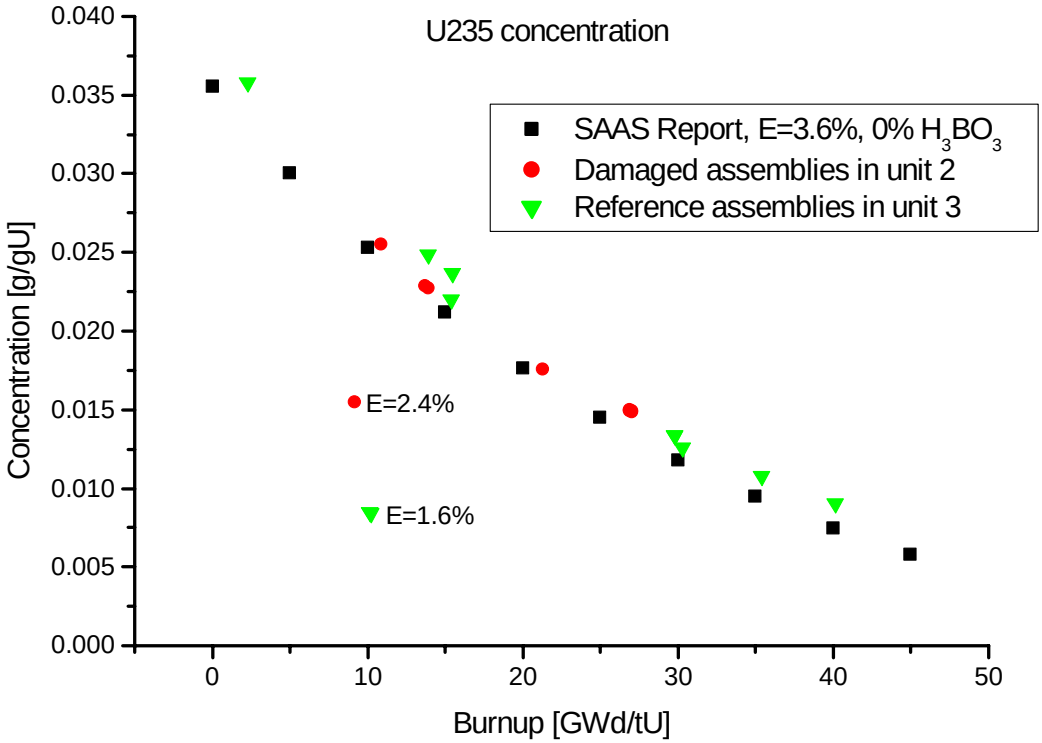


Fig. 5. Concentration of ^{235}U for spent WWER-440 assemblies of 3.6% initial enrichment as a function of burn-up, based on literature data [8] (for 0% boron (H_3BO_3) concentration) and from depletion calculations for the damaged assemblies and reference assemblies at Paks NPP. The outlier points correspond to the single damaged assembly of 2.4% enrichment and to the reference assemblies of 1.6 % enrichment.

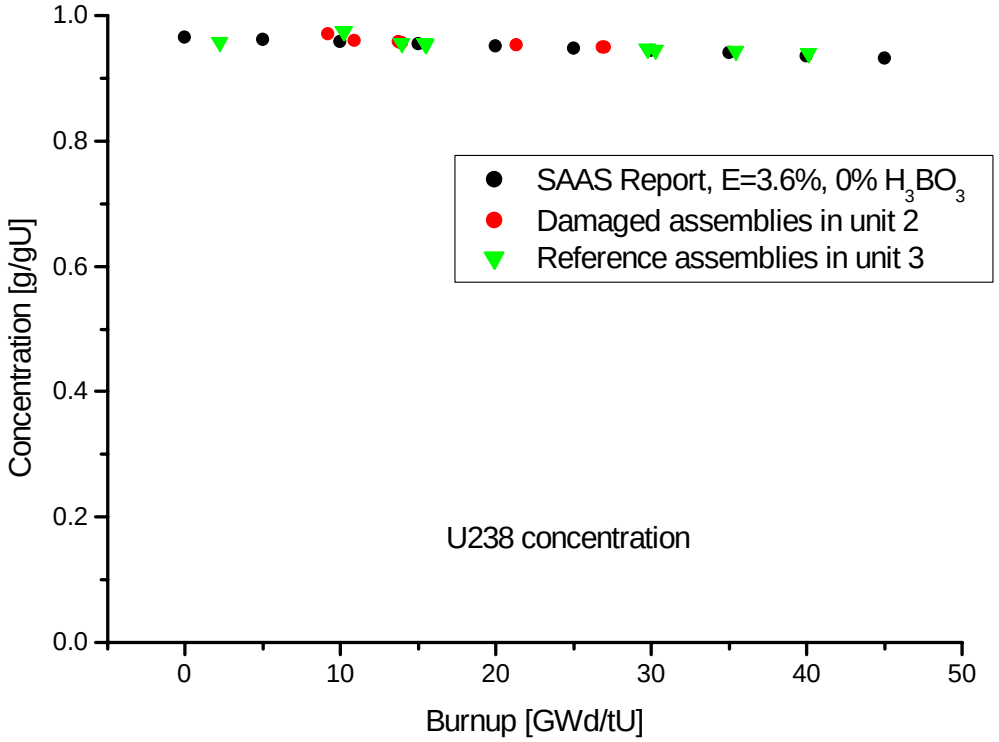


Fig. 6. Concentration of ^{238}U for spent WWER-440 assemblies of 3.6% initial enrichment as a function of burn-up, based literature data [8] (for 0% boron (H_3BO_3) concentration) and from depletion calculations for the damaged assemblies and reference assemblies at Paks NPP.

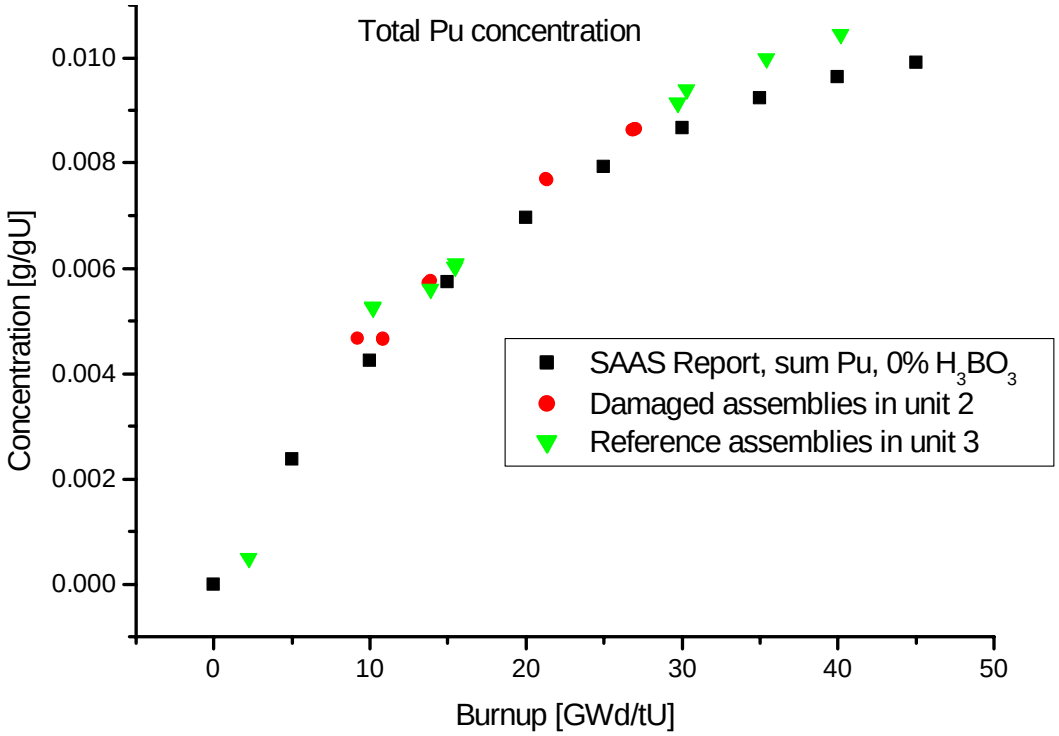


Fig. 7. Concentration of total Pu for spent WWER-440 assemblies of 3.6% initial enrichment as a function of burn-up, based on literature data [8] (for 0% boron (H_3BO_3) concentration) and from depletion calculations for the damaged assemblies and reference assemblies at Paks NPP.

We will verify these correlations by test measurements on regular spent-fuel assemblies in the spent-fuel pond and loading pit. In the burn-up range of interest for us, the concentration of the nuclear material remaining in the spent fuel is calculated using the formulas

$$\rho(U_{total}) = k_0(U_{total}) + k_1(U_{total})BU$$
$$\rho(U235) = k_0(U235) + k_1(U235)BU \quad (3)$$
$$\rho(Pu) = k_1(Pu)BU + k_2(Pu)BU^2$$

where the *k*-s are calibration constants, which are to be determined from depletion calculations verified by measurements of regular fuel assemblies of known burn-up and nuclear material content.

For those containers which will be filled with fuel from a single, identifiable damaged assembly the burn-up is known from the depletion calculations. For those containers, however, which will be filled with fuel parts from various damaged assemblies or with the mixture piled up on the bottom of the cleaning tank the only way to determine the burn-up will be the procedure described in section 5.1, i.e. one should calculate an effective burn-up from the gamma-spectroscopic data. After the effective burn-up is determined, the concentrations of the various fissionable isotopes will be calculated using the above described correlations.

5.3. Determining the mass of the fissionable material

In order to determine the mass of the fissionable material, one needs both the gamma-spectroscopic information and the measured neutron count rates.

The mass of the fissionable material is given by the product of its concentration and the initial mass of total uranium which was present in the observed volume of the fuel, $m_{ini}(U_{total})$:

$$m(U_{total}) = \rho(U_{total})m_{ini}(U_{total}), \quad (4)$$
$$m(U235) = \rho(U235)m_{ini}(U_{total}), \quad (5)$$
$$m(Pu) = \rho(Pu)m_{ini}(U_{total}). \quad (6)$$

The concentrations of the fissionable isotopes will be determined from formula (3) using the effective burn-up obtained from the gamma spectra of the fuel. The initial mass of total uranium within the observed volume, $m_{ini}(U_{total})$, will be calculated from the measured neutron count rate. Namely, since the absorption of neutrons within the fuel is much less than that of the gamma-photons, neutrons carry information about the whole volume of the observed fuel, i.e. the neutron count rate, *N*, is proportional to the total initial mass of nuclear material within the observed volume:

$$N = f(BU)m_{ini}(U_{total}) \tag{7}$$

The factor of proportionality, *f(BU)*, depends on the burn-up. On the other hand, the neutron count rate is related to the burn-up by an empirical relation of the form

$$N = \alpha_0 (BU)^{\beta}, \tag{8}$$

where α_0 and β are calibration constants. Calibrating the system with a regular fuel assembly one obtains from the above equations

$$f(BU) = \frac{\alpha_0 (BU)^{\beta}}{M_{ini}(U_{total})}, \tag{9}$$

where $M_{ini}(U_{total})$ is the total mass of uranium in a fresh fuel assembly. Using these equations and the correlations between the concentration and burn-up, the mass of fissionable material can be given as

$$m(U_{total}) = \left[k_0(U_{total}) + k_1(U_{total})BU_{eff}\right]\frac{N}{\alpha(BU_{eff})^{\beta}}, \tag{10}$$

$$m(U235) = \left[k_0(U235) + k_1(U235)BU_{eff}\right]\frac{N}{\alpha(BU_{eff})^{\beta}}, \tag{11}$$

$$m(Pu) = \left[k_1(Pu)BU + k_2(Pu)(BU_{eff})^2\right]\frac{N}{\alpha(BU_{eff})^{\beta}}, \tag{12}$$

where we have introduced the notation $\alpha = \alpha_0 / M_{ini}(U_{total})$ and BU_{eff} is the effective burn-up calculated from formula 2, using the measured intensities of the Cs-peaks.
The concentrations and neutron cont rates will be determined in several (e.g. 15-20) distinct points along the container, i.e. a profile will be taken, and the total mass of fissionable material within a container will be calculated as the integral of the values obtained in the individual measurement points.

In order to estimate the error of the described method for determining the mass of nuclear material note that for the damaged-fuel containers merely an ***effective*** burn-up can be measured, while, on the other hand, the correlations given in this section are valid for regular fuel assemblies, having well defined burn-up. Therefore, the calculated mass of fissionable material within the damaged-fuel containers will have a systematic error due to using the effective burn-up in the correlations (3) and (8). In this document the occurrence of this error will be called “mixed-burn-up effect”. Results based on Monte Carlo simulations show that the error due to the mixed-burn-up effect is a superposition of a systematic bias (which can be corrected for) and a random deviation. This is demonstrated, e.g., in Fig. 8, which shows the contribution of the mixed-burn-up effect to the relative error of determining the mass of ^{235}U using formula (11). In the calculations it was simulated that pieces of damaged fuel from 30 hypothetical spent fuel assemblies with burn-up in the range 10-30 GWd/tU are distributed into 60 containers. This burn-up range corresponds to typical spent fuel assemblies used in WWER-440 power reactors. The Monte Carlo simulations can be easily repeated with any input data corresponding to any particular combination of damaged assemblies. Each point in Fig. 8 represents the estimated difference between the real nuclear-material content of a hypothetical container and the value which can be obtained from measurement using the correlations given in this section. The random deviation of the mass of ^{235}U, total uranium and plutonium is approximately the same (i.e. in our case not more than ±5%), because the mixed-burn-up effect mainly manifests itself through the neutron measurements, i.e. through the high non-linearity of formula (8), which has the same influence in formulas (10), (11) and (12). The total experimental error will be, of course, much larger than this because it will also contain the error of the calibration and the statistical error of the measurement devices.

The contribution of the mixed burn-up effect to the total experimental error is not very sensitive to the exact burn-up distribution of the damaged assemblies. Therefore the basic conclusion (i.e. that the

contribution of the mixed burn-up effect is much smaller than the total experimental error) is not affected by the particular choice of input data for the Monte Carlo simulations. In any particular case, of course, the use the exact data of the examined assemblies leads to a more reliable estimation of the mixed-burn-up effect. A more detailed study of the mixed-burn-up effect will be presented elsewhere.

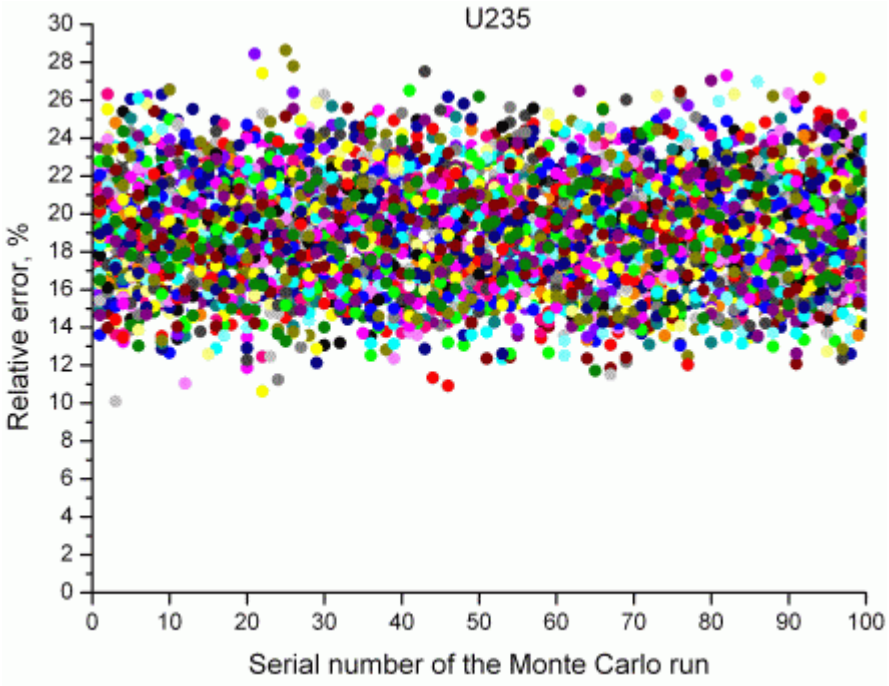


Fig. 8. Contribution of the mixed burn-up effect to the error of determining the mass of ^{235}U.

For comparison, we will also use an alternative method for determining the concentration and mass of Pu, based on measuring the activity of ^{106}Ru. The alternative method relies on the fact that the fission yield of ^{239}Pu is by one order of magnitude larger than the fission yield of ^{235}U, with respect to ^{106}Ru (Table 1). Therefore, the activity of ^{106}Ru is practically proportional to the amount of ^{239}Pu.

5.4. Determining the cooling time

The purpose of determining the cooling time of the damaged fuel, which is, in fact, known, is to verify the accuracy and reliability of the gamma-spectroscopic information. The cooling time can be best determined from the activity ratio ^{144}Ce/^{137}Cs. Namely, the half-life of ^{144}Ce (284.9 days) is much shorter than that of ^{137}Cs (30 years) so the ratio of their activities changes sharply with time. The activity ratio ^{144}Ce/^{137}Cs is determined from the count rate of the 696.5 keV gamma-peak of ^{144}Pr, which is a short-lived (17.29 minutes) daughter element of ^{144}Ce, and from the count rate of the 661.6 keV line of ^{137}Cs. Note, however, that the damaged assemblies have different irradiation histories and they spent different lengths of time outside the reactor, so the “cooling time” of the mixture presently piled-up at the bottom of the cleaning tank is not well defined. Therefore, just as in the case of the burn-up, merely an “effective cooling time” can be measured.

6. Calibration and test measurements

The purpose of the calibration is to obtain the correlations between the following quantities:

- the activity ratios ^{134}Cs/^{137}Cs and ^{154}Eu/^{137}Cs and burn-up
- neutron count rate and burn-up

These correlations are to be obtained using regular spent fuel assemblies. The assemblies for calibration were selected in such a way, that their burn-up covers the full range of burn-ups of the damaged fuel. Their nuclear-material content is known from depletion calculations.

The purpose of the test measurements is, on one hand, to estimate the expected measurement time and the experimental error, and on the other hand to test the reliability of the equipment. The

calibrating and test measurements will be, in fact, the same, the difference being merely in the interpretation.

These measurements will also have a role in determining the smallest measurable amount of nuclear material within a container, relevant to the verification of the nuclear-material-free containers. At the same time they can be used for identifying the possible "centres" where active ^{110m}Ag, present in the welding, may accumulate.

Calibration and testing will be performed in the second phase of the project.

7. Mechanical design of the equipment and experimental layout

The measurements will be performed under water in the spent fuel pond and in the loading pit. The loading pit will be the place of the high-resolution gamma-spectroscopic (and possibly neutron counting) measurements, while in the spent-fuel pond neutron counting and medium-resolution gamma-spectrometric measurements will be performed.

For the measurements in the spent-fuel pond, a dedicated device will be built. The device will contain a pair of fission chambers (type KNK-15), a pair of bubble detectors (BTI BD-PND) and a CZT detector (the exact type is to be decided). The bubbles created in the bubble detectors will automatically be detected and registered as they are created, using acoustic readout[2], without the need to take the detectors out from the device. The measurement geometry will be similar to the geometry applied with the "Enhanced Fork detector" [6], [7], [8]. During the measurements, a few positions in the lattice of the spent-fuel pond around the measurement position will have to be left empty, in order to avoid the disturbing effect of the neighbouring assemblies/containers.

For the high-resolution gamma-spectroscopic measurements the already existing collimator built into the concrete wall of the pit will be used. The high-resolution gamma-spectroscopic measurements of the damaged-fuel containers can be only performed after the recovery process is finished and the cleaning tank is taken away from the loading pit, because at present the cleaning tank blocks the end of the collimator. Therefore, the damaged-fuel containers will have to be taken back from the spent fuel pond to the loading pit to be measured, one by one. This operation will be performed by the refuelling machine. The refuelling machine will also be used to rotate and move the container up and down in front of the collimator, in order that it can be scanned with the HPGe detector along its full length from at least 3 of its sides.

Acknowledgement

Thanks are due to Tamas Biro for useful suggestions and comments on the manuscript.

References

[1] Newsletter of the Hungarian Atomic Energy Authority, February 2003, http://www.haea.gov.hu/web/portal.nsf/irlevelek/C6768704D61BD506C1256DFC006B11AF/$File/hirlevel_2003_majus.doc?OpenElement

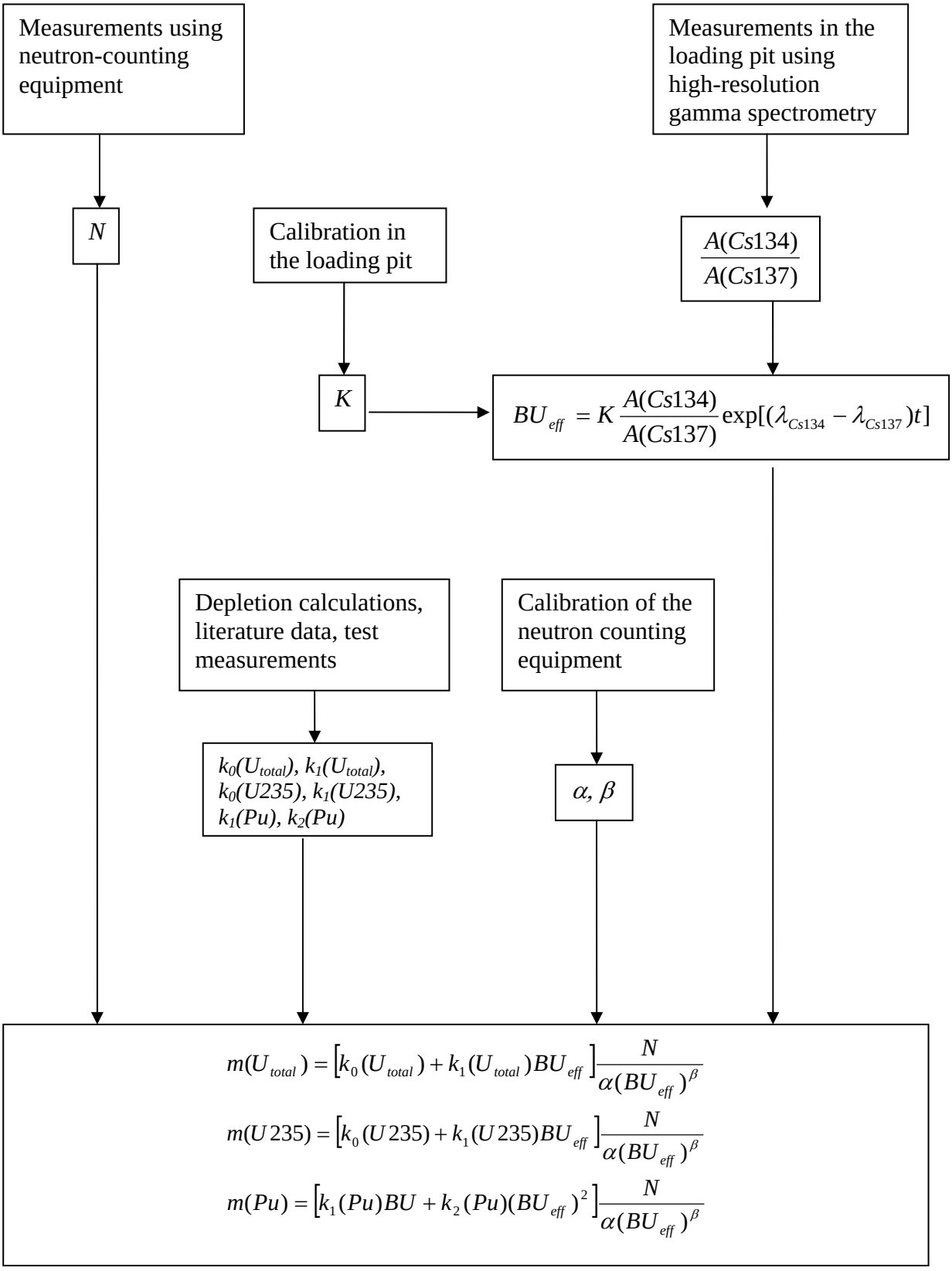
[2] Investigation report on the fuel damage occurred on 10 – 11 April within the Cleaning Equipment of FANP, Report to the Hungarian Atomic Energy Agency – Nuclear Safety Inspectorate, NBI 03. 11. B20301, 2003, http://www.atomeromu.hu/hirek-e/angoljelb20301.doc (see also http://www.atomeromu.hu/hirek-e/bemutato/indexE.html)

[3] Report to the Chairman of the Hungarian Atomic Energy Commission on the Authority's investigation of the incident at Paks Nuclear Power Plant on 10 April 2003 (Identification number of the event: 1120) Hungarian Atomic Energy Authority, 23 May, 2003,

[2] The possibility of the acoustic detection of neutrons by bubble detector was studied previously e.g. in reference [17].

http://www.haea.gov.hu/web/portal.nsf/download/2BBE5BE53B4FA81BC1256DEE004ED9C3/$File/haecreport041003_corrected.pdf?OpenElement
[4] *Nuclear Safety Review for the year 2003*, *IAEA/NSR/2003*, IAEA, Vienna, 2004. http://www-ns.iaea.org/downloads/ni/publications/nsr2003definitive.pdf
[5] Animation on the web-site of Paks NPP: http://www.atomeromu.hu/hirek/bemutato/animacija2.html
[6] A. Tiitta, J. Hautamäki: *Spent VVER fuel characterisation combining a fork detector with gamma spectrometry*. Interim report on Task JNT A1071 FIN of the Finnish Support Programme to IAEA Safeguards. STUK-YTO-TR 181. Helsinki 2001.
[7] A. Tiitta, J. Saarinen, M. Tarvainen, K. Axell, P. Jansson, R. Carchon, J. Gerits, Y. Kulikov, Y.G. Lee: *Investigation on the possibility to use fork detector for partial defect verification of spent LWR fuel assemblies*. Final report on Task JNT A1071 (BEL, FIN, SWE) of the Member States' Support Programme to IAEA Safeguards. STUK-YTO-TR 191. Helsinki 2002.
[8] A. Tiitta, J. Hautamäki, A. Turunen, R. Arlt, J. Arenas Carrasco, K. Esmailpour-Kazerouni, P. Schwalbach: *Spent BWR fuel characterisation combining a fork detector with gamma spectrometry*. Report on Task JNT A 1071 FIN of the Finnish Support Programme to IAEA Safeguards. STUK-YTO-TR 175. Helsinki 2001.
[9] A. Lebrun, M. Merelli, J-L. Szabo, M. Huver, R. Arlt, J. Arenas-Carrasco: SMOPY a new NDA tool for safeguards of LEU and MOX spent fuel, IAEA-SM-367/14/03
[10] S. Szucsán: Preparations for the recovery of the damaged fuel in block 2 of Paks NPP, 3rd Nuclear Technology Symposium, Budapest, Hungarian Nuclear Society, Budapest 2005 (in Hungarian)
[11] A. Aszódi, G. Légrádi: A tisztítótartályban található sérült üzemanyag részletes háromdimenziós termohidraulikai modellezése (Detailed three-dimensional thermo-hydraulic modeling of the damaged fuel in the cleaning tank) , 3rd Nuclear Technology Symposium, Budapest, Hungarian Nuclear Society, Budapest 2005 (in Hungarian)
[12] J. Kópházi, S. Fehér, Sz. Czifrus, T. Berki, Gy. Csom: A tisztítótartályban található sérült üzemanyag részletes háromdimenziós reaktorfizikai modellezése (Detailed three-dimensional reactor-physical modeling of the damaged fuel in the cleaning tank), 3rd Nuclear Technology Symposium, Budapest, Hungarian Nuclear Society, Budapest 2005 (in Hungarian)
[13] A. Cserháti: A paksi sérült üzemanyag eltávolításának technológiája (The technology of the damaged-fuel removal), 3rd Nuclear Technology Symposium, Budapest, Hungarian Nuclear Society, Budapest 2005 (in Hungarian)
[14] L. Lakosi, J. Sáfár, I. Józsa: Spent Fuel Measurements at NPP Paks, ESARDA Bulletin No. 13, 31-32 (1987)
[15] W. Moller, M. Burmester: *Calculation of passive neutron emission from spent WWER-440 fuel*, Report SAAS-347, Staatliches Amt für Atomsicherheit und Strahlenschutz, Berlin, GDR, 1987
[16] C. Tam Nguyen, K. Baricza, I. Pavlicsek and L. Lakosi: *Investigation of fast neutron emission of spent fuel assemblies with CR-39 track etch detector*, Radiation Measurements 25, pp. 695-698 (1995)
[17] T. Bíró, A Kelemen, I. Pavlicsek: Acoustic detection of neutrons by bubble detectors, Nucl. Rad. Meas. **17**/4, 587-589 (1990)

Appendix: Flow chart of the determination of the nuclear-material content of the damaged-fuel containers



An assessment of the FORK detector as a partial defect tester

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Abstract:

A Coordinated Technical Meeting on Spent Fuel Verification Methods held in Vienna in March 2003 stressed the need for partial defect testers for spent fuel in both wet and dry storage.
One of the candidates for partial defect testing is the FORK detector. The performance of the FORK detector as a partial defect tester was investigated experimentally in the framework of a Joint Task for the IAEA by the Finnish, Swedish and Belgian Support Programmes. This was done by replacing fission chambers in the FORK detector by He-3 tubes and simulating spent fuel with fresh MOX-fuel.
It was concluded that the FORK detector was not suitable as a partial defect tester, since fuel pin removal could be not be detected in an unambiguous way for all possible scenarios.

However, later analysis of the results revealed that a possibility exist to interpret the measurement results unambiguously, provided that some additional measurements are performed.
It was decided to perform simulation calculations for some of the configurations that were investigated during the Joint Task for validation of the correctness of the simulation methodology and to investigate the applicability of the proposed additional measurements for an unambiguous partial defect test.

This paper describes the comparison of the simulation calculations with the FORK detector measurements.

Keywords: spent fuel, FORK detector, MCNP

1. Introduction

75% of the nuclear material that is under IAEA safeguards consists of plutonium in spent fuel. In general this type of nuclear material is hard to verify accurately due to the high radiation level of the fission products in spent fuel. At the moment there is not yet a comprehensive method that can function as a partial defect tester for spent fuel assemblies under all circumstances. A Coordinated Technical Meeting on Spent Fuel Verification Methods held in Vienna in March 2003 stressed the need for such partial defect testers for spent fuel in both wet and dry storage.
The FORK detector (FDET) is considered for a long time as a partial defect tester for spent fuel, but there was a need for operator's data for an unambiguous inspection result. Moreover, recently some doubts were raised whether the FDET could detect a diversion of spent fuel pins in all cases. This led to a request from the IAEA to have a tool to investigate in an easy way the capability of the FDET to cover all diversion scenarios. The Belgian Support Programme has accepted this task to develop an easy-to-handle MCNP input file for the FDET and to verify its partial defect testing capabilities.

2. Description of previous experiments

In the framework of the joint IAEA support task JNT A 1071 by the Finnish, Swedish and Belgian Support Programmes, Fork detector (FDET) measurements have been carried out among others in the VENUS critical facility. The aim of these measurements was to simulate diversion scenarios of spent fuel. For this purpose fresh fuel rods were used that consist of a mixture of uranium (2%

enriched) and plutonium (2.7%) fuel. Contrary to spent fuel rods, these fresh fuel rods can be manipulated easily in the VENUS facility so several diversion paths could be investigated.
Neutrons emitted by mainly ^{240}Pu simulated the neutrons coming from ^{242}Cm and ^{244}Cm in spent fuel. Gamma photons from ^{241}Am and several Pu isotopes simulated the gammas from fission products in spent fuel, mainly ^{137}Cs. Since the number of neutrons emitted by ^{240}Pu is orders of magnitude lower than that from the Cm isotopes in spent fuel, the four standard fission chambers in the FDET were replaced by two ^{3}He chambers, which have a much higher sensitivity for neutrons. For the gamma measurements four ionisation chambers were used in stead of the standard two ionisation chambers. The lower gamma energy of the gammas from ^{241}Am compared to gammas from ^{137}Cs was neglected in the simulation. A more extensive description of these experiments is given in [1, 2].
Gamma and neutron total counts were measured for several different fuel configurations with different numbers of fuel pins missing. The fuel configurations discussed in this paper are shown in the annex.

3. Description of input

3.1. Gamma source

The gamma source in the MCNP input consists of a combination of 50 keV from ^{241}Am and several gammas in the few 100 keV range from Pu isotopes. This is simulated by two gamma energy groups, one between 0 and 100 keV and one between 100 and 300 keV. Since the higher energy group contributed almost three orders of magnitude more than the lower group, the gamma problem became effectively a one group problem. Gammas with higher energies did not give a significant contribution to the spectrum.
One can question the usefulness of the simulation of spent fuel by the fresh fuel with respect to the gammas. The energy of gammas from Pu isotopes is significantly lower than the dominant 661 keV gamma from the fission product ^{137}Cs in real spent fuel.

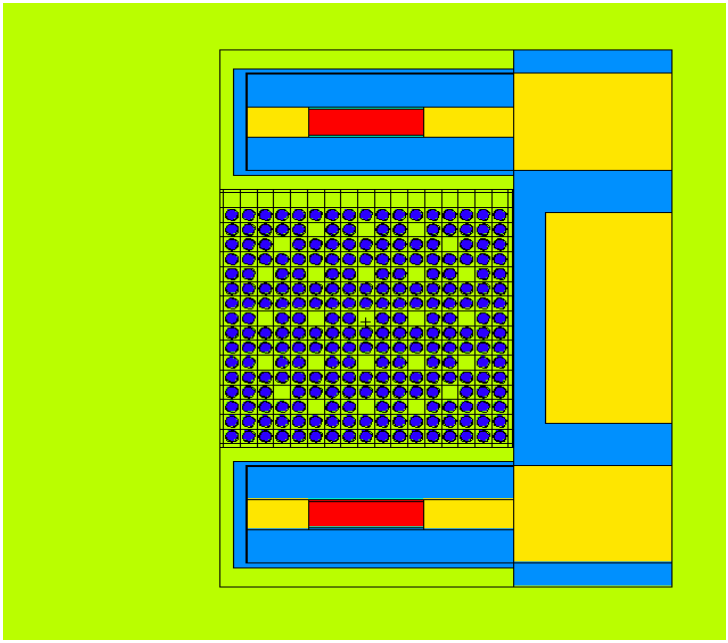

Figure 1a. Horizontal cross-section of the measurement set-up showing the fuel assembly with one fuel column missing

3.2. Neutron source

The neutron source in the MCNP input is neutrons from the spontaneous fission of ^{240}Pu. Parameters for this spectrum can be given as standard input in MCNP.

3.3. General measurement set-up

The general measurement set-up of the FDET and the fuel assembly, as simulated in the MCNP calculations, is shown in figure 1a and 1b. Figure 1a shows a horizontal cross-section of the measurement set-up. A fuel assembly is shown where the fuel column at the right is removed. The position of one detector is visible, the central ionisation chamber.
Figure 1b contains a vertical cross-section of the measurement set-up. In this figure the position of the ionisation and ^{3}He chambers is shown. The ionisation chambers have a larger diameter than the ^{3}He chamber.

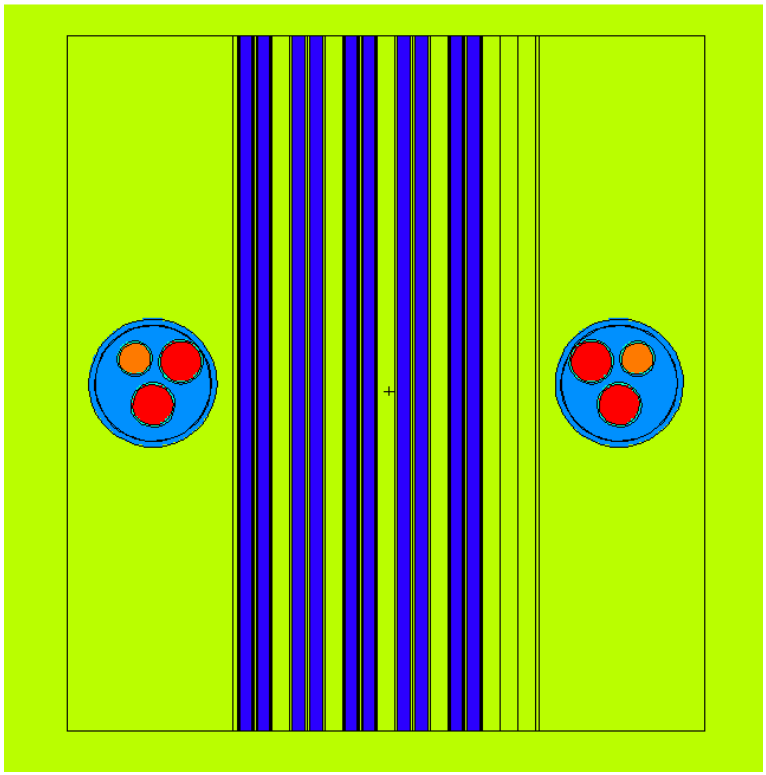

Figure 1b. Vertical cross-section of the measurement set-up showing the positions of the detectors

3.4. Calculations

The calculations have been performed with MCNPX version 2.5.b. Neutron and gamma fluxes have been calculated in separate runs in the cells that contain the filling gases of the different detectors. Weighting factors have been used for optimising the statistics of the calculations for those specific cells. Run times for a typical gamma calculation was 30 minutes for obtaining an uncertainty of 1%. For neutrons the run time was 15 minutes for an uncertainty of 1%.

4. Results of the calculations and discussion

The results that are obtained show that the calculated normalised neutron signal is within 2% of the measured neutron signal. The calculations are performed with sufficient statistics for a 1% uncertainty (1σ) and the measurements with an uncertainty of 0.3% (1σ). Nevertheless, the results are satisfactory for the purpose of the calculation, i.e. to verify the capability of the FDET for partial defect tester.
The calculated gamma signals deviate significantly more from the measured values than the neutrons. The average deviation is 6%. Since the measurement uncertainty was not mentioned (a drift of the signal was reported of about 10% in 2 hours), it is in principle not easy to interpret the calculation results. However, in view of the fact that the measured signals are systematically lower than the calculated values, it can be assumed that the energy of the gammas has been set a little too high in the MCNP calculations.

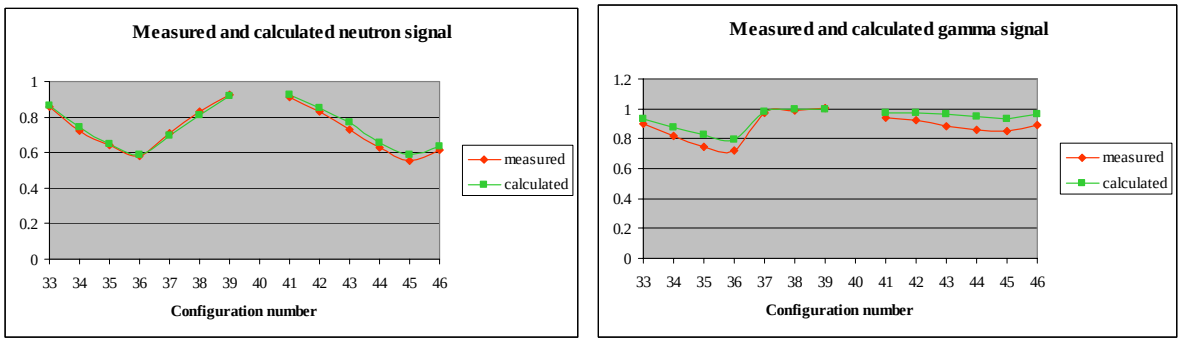

Figure 2. Comparison between the measured and calculated neutron signal (left) and gamma signal (right). Both the measured and calculated values have been normalised to the results of configuration 32, the full assembly.

A common understanding of measurements with the FDET is that the neutron signal is proportional to the amount of fuel in the measured fuel assembly. Figure 3 shows the measured neutron signal and the same signal, but then corrected for the number of pins left in the fuel assembly. Based on the common understanding the corrected signal should be equal to 1, but it is clear that the more fuel is missing, the more the corrected signal deviates from 1. Geometry effects could be a cause, but more plausible is the fact that by diminishing the amount of rods in the assembly also the reactivity of the assembly decreases with a subsequent less multiplication of neutrons. The data in figure 3 are derived from the measurements on fresh fuel. This reactivity effect is more pronounced in fresh fuel than in spent fuel due to the lower content of fissile material in the latter.

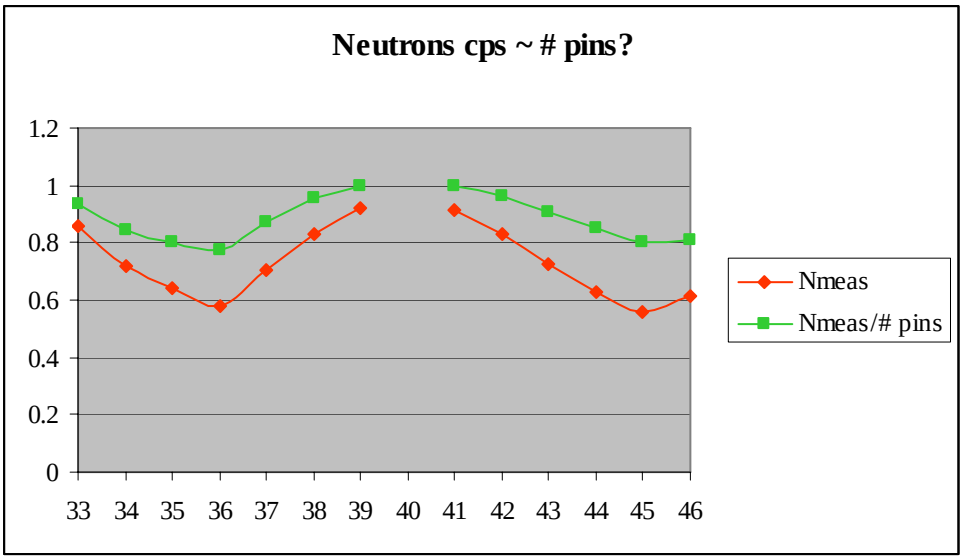

Figure 3. Measured neutron signal (lower line) and the same signal corrected for the number of pins left in the assembly (upper line).

5. Conclusions

MCNPX calculations have shown that it is possible to simulate FDET measurements on fresh MOX rods in the VENUS facility at SCK•CEN. The results show that the gamma energies have to be adapted slightly.

Future work still has to be performed to calculate a significant number of diversion strategies in order to verify whether the FDET is suitable as a partial defect tester for spent fuel assemblies.

6. References

[1] Carchon R., Bruggeman M., Koulikov Y., Lee Y.G., De Baere P., "Measurement of fresh LWR fuel Assemblies with the IAEA FORK Detector", SCK•CEN, Mol, Belgium, Report R-3614, April 2002

[2] Tiitta A., Saarinen J., Tarvainen M., Axell K., Jansson P., Carchon R., Gerits J., Koulikov Y., Lee Y.G., "Investigation on the Possibility to use FORK Detector for Partial Defect Verification Of Spent LWR Fuel Assemblies", STUK, Helsinki, Finland, Report STUK-YTO-TR 191, September 2002

Annex. Description of the different fuel configurations

The fuel configurations that were calculated consist of a standard 17 by 17 PWR fuel assembly with 264 fuel pins and 25 water holes. The positions of the water holes are shown in the scheme below with an **X**.

																A
																B
					X			X			X						C
			X									X				D
																E
		X			X			X			X			X			F
																G
																H
		X			X			X			X			X			I
																J
																K
		X			X			X			X			X			L
																M
			X									X				N
					X			X			X						O
																P
																Q
17	16	15	14	13	12	11	10	9	8	7	6	5	4	3	2	1	

Columns are indicated by a number, rows by a letter. The detectors of the FDET were placed parallel to the columns.
In the investigated configurations fuel rods were removed from either columns or rows. The following table gives the configuration number and which rows or columns were removed in that configuration.

Configuration	removed rows or columns	number of pins left
32	none, full configuration	264
33	17	247
34	17, 16	230
35	17, 16, 15	216
36	17, 16, 15, 14	201
37	16, 15, 14	218
38	15, 14	235
39	14	249
41	A	247
42	A, C	233
43	A, C, E	216
44	A, C, E, G	199
45	A, C, E, G, I	187
46	C, E, G, I	204

Table. Rows (letters) or columns (numbers) that have been removed in the investigated configurations

A semi-empirical technique for verification of spent nuclear fuel assemblies

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Abstract:

The ability to verify operator declared data on spent nuclear fuel is generally considered to be an important issue of safeguards and various techniques have therefore been assigned for this purpose in previous works.

The parameters traditionally used for fuel verification are cooling time and burnup. However, different irradiation histories can result in the same burnup, including those not compatible with civil operation. In this paper, we therefore suggest a method to determine the consistency of the operator declared irradiation history rather than burnup.

The basis of the proposed method constitutes of spectroscopic measurements of the gamma ray emission from various isotopes in the fuel assemblies. The isotopes considered in this work were ^{137}Cs, ^{134}Cs and ^{154}Eu.

The method presented is semi-empirical in the sense that operator-declared information is used to correlate calculated isotopic content with experimentally obtained data.

The method was investigated theoretically in order to determine limits for detecting small deviations from declared data. Also, a preliminary experimental study was performed on PWR fuel assemblies, which is also reported in this paper.

It has been shown that the proposed method makes it possible to discriminate a case where an operator tries to hide a highly irregular irradiation history in order to maximise the production of ^{239}Pu . The cooling time of a fuel assembly may be verified within about 20 days at cooling times extending up to 10-20 years.

Keywords: spent nuclear fuel; verification; irradiation history; cooling time; high-resolution gamma ray spectroscopy

1. Introduction

To merely verify the discharge burnup will not in general reveal the irradiation history of a fuel assembly since many different irradiation histories can result in the same discharge burnup. The proposed method offers a possibility to experimentally verify that a nuclear fuel has been operated as stated with respect to the irradiation history. The method implies that three isotopes, with different production ways and decay rates, are used. Because of that, it is, in principle, very difficult to falsify an irradiation history that can hide an untrue irradiation history. However, various experimental and calculational uncertainties relax that statement somewhat and therefore a theoretical study was performed in order to determine to what extent the method can reveal small deviations between declared and actual irradiation histories.

The main application of the proposed method would probably be in conjunction with long-term storages where assemblies with cooling times up to about 20 years may be investigated. However, it can be foreseen that the method also might be of interest for fuel assemblies with cooling times of only a few years whenever suspect operation of reactors needs to be clarified.

It is assumed that the operator can provide relevant data about the fuel assemblies. If fuel data are not available or incomplete, other methods need to be used in order to experimentally verify the fuel parameters, see e.g. [1].

2. Theory

Irradiation history and cooling time are the most dominant parameters that govern the concentration of fission products in a spent fuel assembly at a given moment.

For a spent fuel with a few years cooling time, the most dominant gamma radiating isotopes are ^{137}Cs, ^{134}Cs and ^{154}Eu, with half lives of about 30 years, 2 years and 8.6 years, respectively. ^{137}Cs is basically produced linearly with respect to burnup while ^{134}Cs and ^{154}Eu are produced essentially quadratically with respect to burnup. ^{134}Cs and ^{154}Eu are dependent on the initial enrichment while ^{137}Cs is not affected by this parameter.

The relations between the calculated activity I, at the time of discharge, and the measured count rate i, for the isotopes considered in this work, can be written as:

$$\begin{cases} i_{137} \cdot e^{\lambda_{137} T} = K_{137} \cdot I_{137} \\ i_{134} \cdot e^{\lambda_{134} T} = K_{134} \cdot I_{134} \\ i_{154} \cdot e^{\lambda_{154} T} = K_{154} \cdot I_{154} \end{cases} \quad (1)$$

where λ_x is the decay constant, K_x is a proportionality constant and T is the cooling time. The left-hand side $i_x \cdot e^{\lambda_x T}$ therefore represents the intensity at the time of discharge. The intensity I_x is among others, governed by the irradiation history provided by the operator and can be calculated in a variety of ways. In this paper, we have illustrated the method using the spent nuclear fuel program, SNF, [2] as shown in section 5. SNF may be used to calculate e.g. isotopic concentration and radiation source terms of spent LWR fuel assemblies. SNF is based on the fuel description and history data generated in core simulation codes such as CASMO-4/SIMULATE-3 and HELIS/PRESTO-2. This means that the information about the isotopic content should, in principle, be available for inspection use.

The proportionality constant K_x of (1) may be determined by using an assembly with known geometry, known irradiation history and with known cooling time. In this work, however, we have adopted the approach of using several fuel assemblies with known properties and fitting a line to each of the expressions in (1). K_x is dependent on the geometry of the specific measurement setup and K_x needs to be re-established every time the measurement geometry is changed. It follows that K_x must also be established for each fuel type encountered by e.g. using a reference assembly or a set of assemblies for each fuel type of interest.

From (1) it is inferred that whenever the irradiation history is correctly declared, the cooling time, as calculated using the various isotopes, should give the same result within the error bars. It is not feasible, however, to solve each equation in (1) separately for T as the sizes of λ_{137} and λ_{154} give excessively large contributions to the uncertainties in the cooling time derivation. Instead, we here utilise the combination of the intensities ^{134}Cs + ^{137}Cs and ^{134}Cs + ^{154}Eu, respectively. From (1) we get the following two expressions

$$T_1 = \frac{1}{\lambda_{137} - \lambda_{134}} \ln\left(\frac{K_{137} \cdot I_{137} \cdot i_{134}}{K_{134} \cdot I_{134} \cdot i_{137}} \right)$$
$$T_2 = \frac{1}{\lambda_{154} - \lambda_{134}} \ln\left(\frac{K_{154} \cdot I_{154} \cdot i_{134}}{K_{134} \cdot I_{134} \cdot i_{154}} \right) \quad (2)$$

It may now be argued that if the irradiation history is erroneously declared, the following expression holds

$$|T_1 - T_2| > \sqrt{(\Delta T_1)^2 + (\Delta T_2)^2} \quad (3)$$

where ΔT_1 and ΔT_2 are the uncertainties of T_1 and T_2, respectively.

2.1. Accuracy considerations

The accuracy of the method is mainly dependent on: (i) the measuring uncertainty of the intensity i_x and (ii) the accuracy with which the activity I_x is calculated.

(i) From the spectral analysis of the experimental study reported in section 5, the relative statistical uncertainties of the measured intensities from i_{137}, i_{134} and i_{154} were estimated to be 0.05%, 0.15% and 0.15%, respectively, for fuels with cooling times around 10 years.

(ii) A measure of the achievable accuracy of I_x was obtained from the results of the experimental study by determining the relative uncertainty of K_x. Omitting the small uncertainty of i_x, we find that the experimental and calculated intensities correlates within 1.3%, 1.1% and 0.9% for ^{137}Cs, ^{134}Cs and ^{154}Eu, respectively.

An error analysis on experimental data showed that the error of T_1 and T_2 was 9 days and 22 days, respectively. According to (3), the difference should be bigger than 24 days or $|T_1 - T_2|$ > 24 days. If this difference deviates significantly from 24 days, one may therefore expect the irradiation history to be erroneously declared.

2.2. A theoretical study

In this section a study is presented where a few simulated irradiation histories have been investigated in order to estimate the detection limits of the technique. Obviously, there is a large amount of conceivable scenarios where various irradiation histories are incorrectly declared. However, the ones chosen for this work represents two distinct cases where:

1. An operator deliberately declares a fuel assembly incorrectly in order to cover illegal activities, such as operating the reactor in a way to maximize the production of ^{239}Pu.
2. Unintentional errors have been made in the bookkeeping of a fuel assembly.

2.2.1. Case 1

This case treats the following scenario: An investigated 15x15 PWR fuel assembly with an initial enrichment of 3.1% has been declared as being irradiated according to table 2a and with a cooling time of 10 years. In table 2a, where no burnup is indicated this relates to the yearly maintenance shutdown of the reactor.

Irradiation history [days]	Burnup [GWd/tU]
335	5
30	
335	5
30	
335	5
Sum	**15**

Table 2a: Declared irradiation history (cooling time 10 years)

Irradiation history [days]	Burnup [GWd/tU]
180	1
180	
330	14
Sum	**15**

Table 2b: Actual irradiation history (cooling time 10 years)

In reality, the fuel assembly was operated in such a way as to maximize the production of ^{239}Pu. This was achieved by irradiating the assembly for 6 months at a low neutron flux that resulted in a burnup of 1 GWd/tU. To cover this particular irradiation history, the assembly was taken out for 6 months and then put back into operation again where it was irradiated for one additional year to obtain a final burnup of 15 GWd/tU according to table 2b.

This case thus represents a scenario where the verification of only burnup is not sufficient to reveal the highly irregular irradiation history of the fuel assembly.

With these assumptions, (2) gives the following calculated cooling times. The uncertainties of T_1 and T_2 are also denoted in table 3.

Cooling time	T_x [years ± days]
T_1	9.6 ± 9
T_2	9.4 ± 22

Table 3: Calculated cooling times using (2) together with the associated uncertainties.

The difference, $|T_1 - T_2|$ = 57 days is about 2.5 standard deviations from expected and gives a reasonably good indication that the irradiation history is not correctly declared. It could be noted that this conclusion is not dependent on the actual cooling time.

2.2.2. Case 2

In this case the following scenario is treated: A measured 15x15 PWR-fuel assembly with initial enrichment of 3.1% has been declared as being irradiated for 6 cycles, each lasting for 335 days followed by maintenance periods of 30 days. The final burnup was 30 GWd/tU evenly distributed among the power cycles and the declared cooling time was 10 years. This simple case treats the situation where the actual cooling time differs ±1 year from the declared value while the irradiation history is correctly declared. The result can be seen in table 4.

$T_{declared}$ **11 years**	T_x **[years±days]**	**Deviation from actual cooling time [days]**
T_1	10 ± 9	0
T_2	10 ± 22	0
9 years		
T_1	10 ± 9	0
T_2	10 ± 22	0

Table 4: Calculated cooling times when the irradiation history is correctly declared while the declared cooling time differs ±1 year.

As shown in table 4, the calculated cooling times T_1 and T_2 are not dependent on declared cooling time, which implies that the correct cooling time is deduced whenever the irradiation history is correctly declared.

2.3. Conclusion

According to the theoretical investigation, the proposed method has a potential to reveal, within reasonable limits, erroneous declarations of irradiation history and cooling time. The method is especially sensitive to incorrectly declared cooling times where deviations of 10 to 20 days may be detected.

From the discussion above it follows that the case where both cooling time and irradiation history are incorrectly declared may also be covered.

3. Mechanical arrangement of the experimental study

The measurements made in the preliminary, experimental investigation have been performed using the gamma scanning equipment installed in all Swedish nuclear power plants and the Swedish interim storage facility CLAB. The system consists of a collimator, an elevator with control system and fixtures for holding the fuel assembly in a vertical position in the elevator. On the collimator, a fixture for holding the detector is mounted. The fuel to be measured is placed in the fixture and rotated along its axis so that one corner is facing the detector. For a detailed account of the measurement equipment see e.g. [3].

The speed of the elevator was set to about 120 cm per minute while travelling upwards and about 150 cm per minute while travelling downwards. The speeds were considered a reasonable compromise in order to minimize the measurement time while keeping uncertainties due to counting statistics on a reasonably low level.

4. Analysis of spectra

The fuel assemblies were scanned with one corner at the time facing the detector [1] and the measurements of each corner of the fuel assembly were divided into 210 subspectra, each representing a fuel length of about 2 cm and corresponding to a measuring time of about 1 second.

During measurements, the high counting rates encountered (up to 100 000 cps) vary significantly and require a proper dead time correction of the spectra. Each subspectrum was therefore corrected using the pulser method where pulses are injected into the detector system from an external pulse generator with a well-defined event rate. By forming the ratio between the number of pulses injected during the measuring time for each subspectrum and the actual number of counts in the pulser peak in the spectrum, a dead time correction factor was obtained for each subspectrum. The content of each channel of a subspectrum was then multiplied with the corresponding dead time correction factor in

order to get properly dead time corrected spectra, Finally, all subspectra were added yielding a total spectrum, see figure 1, of each corner of a fuel assembly from which the net count rates of ^{137}Cs, ^{134}Cs and ^{154}Eu were extracted.

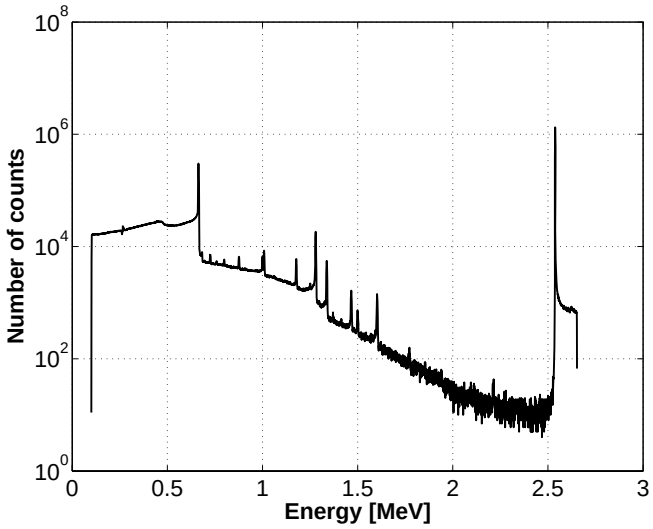

Figure 1: A typical spectrum of one corner of a fuel assembly with a burnup of around 30 GWd/tU and a cooling time of 17 years. The peak around 2.5 MeV is the pulser peak used for dead time correction.

The peak around 0.6 MeV in figure 1 emanates from ^{137}Cs while the small structure at 0.8 MeV corresponds to ^{134}Cs. The intensity corresponding to ^{154}Eu can be found 1.275 MeV and the large peak at 2.5 MeV is the pulser peak. The intensities used in conjunction with (2) represent the sum of intensities from each corner of a fuel assembly. In such a way possible variations in the radial burnup distributions are minimised.

In figure 2, a typical axial profile of ^{137}Cs, is shown for a scanned 14x14 PWR-fuel. Each point in the figures corresponds to the evaluation of one subspectrum. The spacers of the fuel are clearly seen as dips in the intensity distribution. As the fuel assembly is elevated, the top of the fuel is what the detector first sees, this is illustrated as a rise in intensity at an axial position of about 70. A slightly higher burnup can be seen at the bottom part of the fuel.

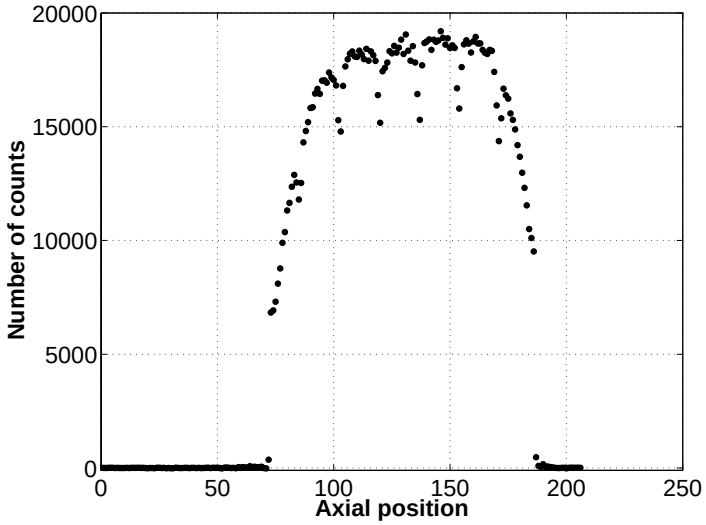

Figure 2: The axial profile of ^{137}Cs. The bottom of the fuel is to right in the figure.

5. Experimental results

5.1. Irradiation history verification via the calculated cooling time

5.1.2. Measurements on 17x17 PWR fuel from Ringhals 3

The PWR fuel assemblies that were used to test the method experimentally, all came from the Swedish nuclear power plant Ringhals 3. Altogether, 12 PWR fuel assemblies were gamma scanned and evaluated with respect to the intensities of ^{137}Cs, ^{134}Cs and ^{154}Eu. Table 5 lists some of the fuel properties.

Fuel ID	Enrichment [%]	Burnup [GWd/tU]	Declared T [days]
0C9	3.101	38.45	3968
1C2	3.101	33.31	3972
1C5	3.101	38.48	3967
2C2	3.101	36.59	3968
3C1	3.101	36.58	3968
3C5	3.101	38.37	3968
3C9	3.101	36.56	3968
4C4	3.101	33.32	3972
4C7	3.101	38.36	3968
0E2	3.103	41.63	3196
0E6	3.103	36.00	3198
1E5	3.103	34.64	3196

Table 5: It can be noticed that ten of the fuel assemblies had similar irradiation cycles (four successive cycles) and two fuel assemblies had three successive irradiation cycles (0E6, 1E5). The assemblies were scanned in April 1997.

The fuels were then evaluated according to the equations for T_1 and T_2 and the difference $|T_1 - T_2|$ was formed and compared to the measurement uncertainty of 24 days as described in section 3. As seen in table 6, five fuels deviate from the stated 24 days, but they are all within 1.5σ.

Fuel ID	T_1 [days]	T_2 [days]	$\lvert T_1 - T_2 \rvert$ [days]	Deviating
0C9	3976	3961	16	
1C2	3974	4004	30	Yes
1C5	3965	3943	22	
2C2	3966	3965	1	
3C1	3968	3970	2	
3C5	3957	3943	14	
3C9	3967	3958	10	
4C4	3956	3987	31	Yes
4C7	3967	3940	27	Yes
0E2	3216	3182	34	Yes
0E6	3200	3220	20	
1E5	3197	3233	36	Yes

Table 6: The table shows the calculated cooling times using the equations as described in section 3. The difference is also tabulated with indication whether the difference is bigger than the mentioned 24 days of section 3. It should be noted that for the deviating fuel assemblies the difference is still within 1.5σ.

6. Discussion

The detection limits of the method depend strongly on the accuracy for which the isotopic content can be calculated. Also, a non uniform radial burnup distribution will affect the conclusion. The experimental investigation has shown that the variation of this distribution among the assemblies studied is in the order of 5%. This effect has not yet been incorporated into the analysis and it is therefore expected that the precision of the method will increase when explicitly taking the radial burnup distribution into account.

The case where an operator tries to hide a highly irregular irradiation history in order to maximize the production of ^{239}Pu may be revealed. The cooling time of a fuel assembly may be verified within about 20 days at cooling times extending up to 10-20 years.

7. Acknowledgement

The authors would like to thank the staff at CLAB for good cooperation and valuable assistance during the experiments. Sigurd Børresen is also acknowledged for his help with the SNF data. This work was kindly financed by the Swedish Nuclear Power Inspectorate (SKI).

8. References

[1] M. Tarvainen, A. Bäcklin, and A. Håkansson. *Calibration of the TVO spent BWR reference fuel assembly.* STUK-YTO-TR 37, Finnish Centre for Radiation and Nuclear Safety (STUK), 1992.

[2] S. Børresen, T. Bahadir, M. Kruners. *Validation of CMS/SNF Calculations against Preliminary CLAB Decay Heat Measurements.* ANS Transactions Vol. 91, 2004 Winter Meeting, Washington, D.C., Nov. 14-18, 2004.

[3] P. Jansson. *Studies of Nuclear Fuel by Means of Nuclear Spectroscopic Methods*. PhD thesis, Department of Radiation Sciences, Uppsala University, Uppsala, Sweden, 2002.

Numerical calibration approach for the determination of ^{235}U enrichment in uranium oxide pellets by quantitative NaI(Tl) gamma spectrometry

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Abstract:

The determination of the ^{235}U enrichment can be easily performed for bulk materials presenting an infinite (or near infinite) thickness to a collimated detector. In this situation, the count rate observed for example in the 185.7 keV photopeak of ^{235}U is directly proportional to the concentration of ^{235}U in the volume "seen" by the detector. Experimental calibration of NaI(Tl) detectors can therefore be easily performed with thick reference standards of different enrichments all presenting an infinite thickness, thus generating a calibration independent of geometry of the measured sample. These can then be used for the determination of the enrichment of unknown samples. This is an approach commonly taken by the International Atomic Energy Agency (IAEA) when determining the ^{235}U enrichment of bulk samples.
In the case of small samples, like fuel pellets, the infinite thickness approach does not hold. The observed count rate in the 185.7 keV peak of ^{235}U becomes a function of enrichment, pellet geometry and relative position to the detector. A pure experimental approach would therefore require having reference materials identical in physical properties (size, density, enrichment range...) to the sample being measured. Given the diversity of pellet types encountered in facilities around the world, such an approach is not viable for the IAEA.
In the present work we present an alternative approach involving numerical simulation to predict the response of a NaI(Tl) detector used to measure a set of unknown UO_2 pellets in a controlled geometry. This work aims to demonstrate the feasibility of using solely NDA techniques during verification measurements on pellet oxide materials. It also illustrates the current capability of the IAEA in performing specific numerical calibrations for quantitative NDA assays.

Keywords Monte Carlo, Enrichment, NDA, Calibration, Pellet

1. Proposed Approach

1.1. Basic Principles

Monte Carlo techniques originate from the concept that if a physical system can be described in a complete way in terms of the probabilities of occurrence of all possible events, random sampling of these probability distributions can describe in a statistical way the physical behaviour of the system. Common examples of such systems are games of luck for which, because of their limited complexity, all probabilities of occurrence of winning and loosing combinations can be easily computed.

Monte Carlo techniques cover a wide range of applications but they are of great interest in complex systems where no explicit analytical mathematical solution can easily be expressed/calculated but where each simple individual event can be well characterised and calculated.
The Monte Carlo N-Particle code [1] (MCNP) developed over 50 years at Los Alamos National Laboratory uses this technique to simulate the distribution of path followed by ionising radiation when travelling through matter. The radiation transport is performed by randomly sampling distributions representative of the probability of interaction of the radiation with the material(s) they travel through. The probabilities of interaction are derived from the

geometrical and physical specification of the materials surrounding the source of radiation (chemical composition, density...) and a vast amount of evaluated nuclear data specifying the probability of interaction of radiation with matter for all types of individual physical processes (scattering, absorption...). Using a well conceived random number generator, a large number of source particle can hence be transported thus building a global picture of their fate. This numerical experiment allows tallying specific information representative of events of interest. Of course these results are obtained with an associated statistical uncertainty based on the efficiency of the tallying.

The accuracy of the numerical simulation can be determined by modelling the measurement of well-characterised material (reference standards) in a well-controlled geometry and measuring these standards in a laboratory. The comparison of the two results for various geometry and sample characteristics yields estimations of a benchmark coefficient, or bias correction factor, which if consistent for all the measurement situations can be applied as a systematic error correction to other numerical results in order to predict experimental results. The source of this bias has multiple origins. The main sources are the cumulative effect of the uncertainties in the description of individual physical interactions between radiation and matter (nuclear data), the representation of the geometry and physical characteristics of the modelled objects/materials, and the incomplete representation of physical effects taking place in the real measurements (i.e. electronic noise, charge collection losses, secondary particle modelling...). This represents an intrinsic limitation of these numerical methods. In the case of the modelling of detection processes in HPGe detectors these effects are somewhat limited to discrepancies of the order of a few % while because of the non-modelling of physical effects in the detection/counting process of scintillating detectors this value usually increases to 20-30% for NaI(Tl). This situation is even worse for CdZnTe detectors for which the size of the active volume of the crystal is not accurately known.

In the case of the measurement of LEU fuel pellets, MCNP was used to model the response of a NaI(Tl) detector when measuring a specific set of reference pellets in a known measurement geometry. Note that the modelled reference materials do not have to be fully representative (size, density...) of the specific type of pellets to be later measured during inspection. These initial Monte Carlo calculations produced reference numerical results for the computational method. The benchmark of the numerical reference results, in other words the evaluation of the bias observed between modelled and observed responses for a specific detector, is obtained by experimentally measuring the same limited set of reference material at the IAEA Safeguards Instrumentation Laboratory (SIL).

Once the benchmark/validation of the developed Monte Carlo model has been performed its results could be reliably corrected to predict the real response of the NaI(Tl) detector when measuring (in the same geometry) any type of pellets specified in the model. Hence, varying the ^{235}U enrichment in the specification of the modelled pellets permitted the empirical construction of the relationship between detector response and pellet enrichment. Provided the physical properties of the measured pellets are made available, specific calibration curves for these pellets can therefore be computed in advance of the real measurements, thus allowing when performing the measurements at the inspected facility, the direct interpretation of the measured count rates in terms of ^{235}U enrichment.

2. Work Performed

2.1. Monte Carlo Modelling

A reference model of a NaI(Tl) detector from SCIONIX of the type 51BS12.7/ 2E3-XamT [2] has been developed using the Monte Carlo N Particle code version 5. The detector was modelled inside a Lead Shield of the type CANBERRA 7419BS [3]. The front collimator insert of the type 7419-BS was removed to not limit the field of view of the detector. The detector was modelled with a 1-mm thick Cd absorber on top of the entrance window, which comes level with the lead ring. An aluminium/PVC holder was modelled on top of the detector/lead shield assembly. Figure 1 shows a 2-D vertical cross section of the constructed geometrical model.

In order to limit discrepancies between field and benchmark measurements fixed dimension Aluminium/PVC inserts allowing reproducible pellet positioning have been prepared. Type A inserts allow up to 5 pellets to be positioned above the detector (15-mm holes) while Type B inserts allow for 4 pellets to be measured (20-mm hole). A photograph of the measurement system with a Type A insert is shown in Figure 2. This allows the reliable estimate of the bias between computational and experimental results (for a given detector)

derived at SIL to be identical to the one encountered during inspection.

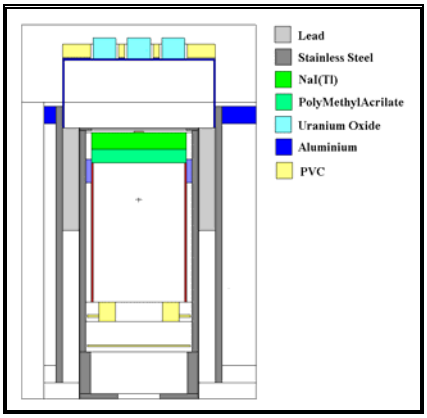

Figure 1: 2-D vertical cross section of the MCNP model showing the NaI(Tl) detector inserted inside the lead shield.

Figure 2: Photography of the NaI(Tl) detector inserted in the Canberra 7419BS Shield with a type A pellet holder.

3. Monte Carlo Model Benchmark

No attempt in deriving an optimal MCNP model of the Scionix probe was undertaken. A series of reference measurements with several single energy point source(s) placed at several positions around the detector could lead to a much better validation of the MCNP model. Such an approach is taken in commercially available software such as the ISOCS [4].
Also note that unless a check-source of known activity is measured in a controlled geometry in the field prior to the measurement of unknowns no direct verification of the validity of the benchmark coefficient derived at the characterisation time is possible. It may therefore be of interest in the future to establish a benchmark validation measurement scheme to ensure the validity of this coefficient.

3.1 Reference Materials

Two sets of reference materials were prepared to perform the benchmark measurements. A set of 15 PWR pellets (SU-106) and a set of 15 BWR pellets (SU-109) were individually packed in protective material (PVC wrapper).

3.1.1 PWR Pellets

Table 1 list the characteristics of the SU-106 Samples.

Density of UO_2 Pellets

ID	Ø mm	length mm	Vol [cm³]	weight [g]	density [g/cm³]	U-235 W-%	U-235* [g]
PWR-E1/21	9.29	15.27	1.035	10.5624	10.205	0.7170%	0.06675
PWR-E1/22	9.28	15.46	1.046	10.6734	10.207	0.7170%	0.067452
PWR-E1/23	9.28	15.44	1.044	10.6843	10.231	0.7170%	0.067521
PWR-E1/24	9.27	15.27	1.031	10.5754	10.261	0.7170%	0.066832
PWR-E1/32	9.27	15.38	1.038	10.6623	10.272	0.7170%	0.067382
<Mean Value>	**0.928**	**1.536**	**1.039**	**10.632**	**10.235**		
Standard Deviation	**0.001**	**0.009**	**0.006**	**0.058**	**0.031**		
R.S.D	**(0.09%)**	**(0.59%)**	**(0.61%)**	**(0.54%)**	**(0.30%)**		
PWR-E2/12	9.27	15.11	1.020	10.5320	10.328	2.5370%	0.235508
PWR-E2/31	9.28	14.49	0.980	10.0946	10.300	2.5370%	0.225727
PWR-E2/32	9.27	14.65	0.989	10.2459	10.362	2.5370%	0.229109
PWR-E2/33	9.27	14.91	1.006	10.4060	10.341	2.5370%	0.23269
PWR-E2/34	9.27	14.83	1.001	10.3510	10.342	2.5370%	0.23146
<Mean Value>	**0.927**	**1.480**	**0.999**	**10.326**	**10.335**		
Standard Deviation	**0.000**	**0.024**	**0.015**	**0.165**	**0.023**		
R.S.D	**(0.05%)**	**(1.61%)**	**(1.54%)**	**(1.60%)**	**(0.22%)**		
PWR-E3/21	9.27	14.73	0.994	10.2697	10.330	3.4930%	0.316175
PWR-E3/22	9.27	14.71	0.993	10.2915	10.366	3.4930%	0.316846
PWR-E3/23	9.27	14.58	0.984	10.1944	10.360	3.4930%	0.313857
PWR-E3/24	9.27	14.75	0.996	10.3632	10.410	3.4930%	0.319056
PWR-E3/32	9.27	14.76	0.996	10.3167	10.356	3.4930%	0.317622
<Mean Value>	**0.927**	**1.471**	**0.993**	**10.287**	**10.364**		
Standard Deviation	**0.000**	**0.007**	**0.005**	**0.062**	**0.029**		
R.S.D	**(0.00%)**	**(0.50%)**	**(0.50%)**	**(0.61%)**	**(0.28%)**		

*: Assuming 88.14% U in UO_2 — **Total U-235 3.073988** g

Table 1 Detailed Characteristics of the reference PWR Pellets used for the MCNP benchmark

4 pellets of similar enrichments were measured on a Type A insert and net measured count rates in the 185.7 keV region of the spectrum was compared with the MCNP derived count rates based on the calculated absolute detection efficiency and emission rates of the pellets at 185.7 keV.

3.1.1 BWR Pellets

Table 2 list the characteristics of the SU-109 Samples.

Density of UO_2 Pellets

ID	Ø mm	length mm	Vol [cm³]	weight [g]	density [g/cm³]	U-235 W-%	U-235* [g]
BWR-E1-26	10.60	12.20	1.077	11.0417	10.256	0.7140%	0.069488
BWR-E1/25	10.59	12.10	1.066	10.9260	10.252	0.7140%	0.068759
BWR-E1/27	10.60	12.24	1.080	11.0874	10.265	0.7140%	0.069775
BWR-E1/27	10.59	12.14	1.069	10.9685	10.256	0.7140%	0.069014
BWR-E1/46	10.61	12.28	1.086	11.1299	10.251	0.7140%	0.070043
<Mean Value>	**1.060**	**1.219**	**1.076**	**11.030**	**10.256**		
Standard Deviation	**0.001**	**0.007**	**0.008**	**0.084**	**0.005**		
R.S.D	**(0.08%)**	**(0.60%)**	**(0.75%)**	**(0.76%)**	**(0.05%)**		
BWR-E2/27	10.61	12.20	1.079	10.9809	10.180	2.0100%	0.194539
BWR-E2/47	10.60	12.16	1.073	10.9381	10.193	2.0100%	0.193781
BWR-E2/48	10.61	12.22	1.080	11.0226	10.202	2.0100%	0.195278
BWR-E2/50	10.61	12.25	1.083	11.0391	10.192	2.0100%	0.19557
BWR-E2-49	10.60	12.23	1.079	11.0417	10.231	2.0100%	0.195616
<Mean Value>	**1.061**	**1.221**	**1.079**	**11.004**	**10.200**		
Standard Deviation	**0.001**	**0.003**	**0.004**	**0.044**	**0.019**		
R.S.D	**(0.05%)**	**(0.28%)**	**(0.34%)**	**(0.40%)**	**(0.19%)**		
BWR-E3-31	10.59	12.16	1.071	11.0043	10.274	3.0500%	0.295825
BWR-E3/42	10.59	12.05	1.061	10.9306	10.299	3.0500%	0.293844
BWR-E3/29	10.61	12.23	1.081	11.1076	10.272	3.0500%	0.298602
BWR-E3/32	10.60	12.08	1.066	10.9094	10.234	3.0500%	0.293274
BWR-E3/30	10.60	11.88	1.048	10.6582	10.166	3.0500%	0.286521
<Mean Value>	**1.060**	**1.208**	**1.066**	**10.922**	**10.249**		
Standard Deviation	**0.001**	**0.013**	**0.012**	**0.167**	**0.052**		
R.S.D	**(0.08%)**	**(1.09%)**	**(1.14%)**	**(1.53%)**	**(0.50%)**		

*: Assuming 88.14% U in UO_2 — **Total U-235 2.789931** g

Table 2 Detailed Characteristics of the reference BWR Pellets used for the MCNP benchmark

		IMCA "Area"		Genie 2000 Peak Area	
Pellet Type	MCNP Rate [s^{-1}]	Experimental Rate [s^{-1}]	Benchmark Factor	Experimental Rate [s^{-1}]	Benchmark Factor
PWR-E1	14.29 ± 0.07	11.38 ± 0.23	**125.54% ± 2.05%**	12.86 ± 0.18	**111.09% ± 1.47%**
PWR-E2	50.36 ± 0.19	37.77 ± 0.52	**133.31% ± 1.43%**	40.78 ± 0.56	**123.49% ± 1.42%**
PWR-E3	69.39 ± 0.25	50.32 ± 0.56	**137.91% ± 1.17%**	54.56 ± 0.74	**127.19% ± 1.41%**
BWR-E1	16.92 ± 0.07	13.13 ± 0.22	**128.93% ± 1.70%**	14.96 ± 0.18	**113.16% ± 1.25%**
BWR-E2	47.90 ± 0.11	33.46 ± 0.24	**143.15% ± 0.75%**	36.62 ± 0.25	**130.78% ± 0.71%**
BWR-E3	72.71 ± 0.25	53.09 ± 0.36	**136.94% ± 0.76%**	60.04 ± 0.47	**121.09% ± 0.85%**
		Mean Benchmark	**134.30% ± 2.61%**	Mean Benchmark	**121.13% ± 3.16%**

Table 3 Summary Results of the Benchmark Measurements

A similar benchmark procedure was used for the BWR pellets.

3.2 Benchmark Results

This section illustrates the results obtained when benchmarking the MCNP predicted rates against series of reference material measurements. Two methods were used to determine the experimental peak areas (see 5.2 Data Analysis) of the 185.7 keV peak. The first was based on a direct reading of the area information displayed by the IMCA software [5], while the second was based in a Canberra Genie 2000 software [6] peak area analysis for NaI(Tl) spectra.
Performing a re-analysis of the experimental count rates using the calculated benchmark to derive an enrichment calibration coefficient lead to measured enrichment values consistent with a 5-% standard deviation with the declared values. Figure 3 displays visually these results.

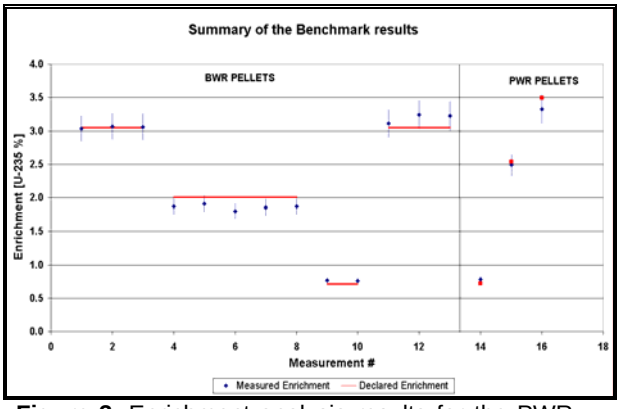

Figure 3: Enrichment analysis results for the PWR and BWR reference pellets based on the Genie 2000 benchmark coefficient.

4. Automation of the Calibration Process

The developed reference model was used as a basic template for the generation of the numerical calibration inputs. Indeed, in order to be able to generate quickly a large number of input files necessary to produce the calibration data, a Microsoft Visual Basic 6.0 application has been created to modify automatically the MCNP cards of a basic template.

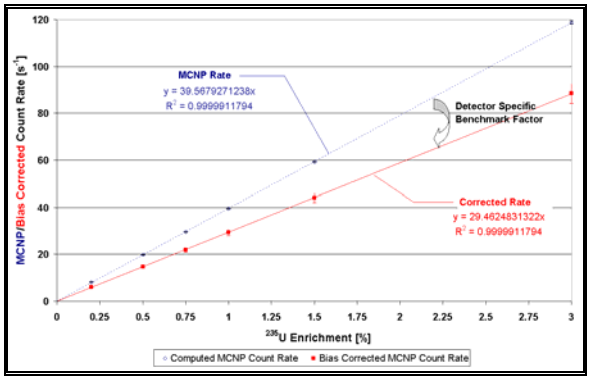

Figure 4: Example of MCNP™ and bias corrected count rates for the derivation of pellet type specific enrichment calibration coefficients.

The user interface of this application allows specifying all the parameters describing the specific characteristics of the measured pellet(s) (pellet height and diameter, density, enrichment...) as well as additional inputs related to the assay itself. This application can be run to call MCNP and perform the Monte Carlo calculation as well as retrieving the uncorrected detector count rates based on the numerical simulation.

Results computed for several enrichments for a given pellets type can then be compiled as a calibration curve which linear slope has to be bias corrected to derive the true calibration coefficient. This is illustrated in Figure 4.

5. Experimental Work

Experimental work in support of this numerical calibration approach for safeguards verification purpose at fuel fabrication facilities was undertaken during the summer of 2004. 36 pellet type calibrations were performed at IAEA Headquarters prior to measurements to cover all types of pellet manufactured in two facilities. This encompased green pellets, unground sintered and ground sintered pellets for several types of nuclear reactors.

5.1 Measurement Set-up

Groups of five pellets from different types of pellets placed on Type A inserts were measured on-site for 900 seconds. Similar counting time were used to initially evaluate the ambient background. The standard IMCA software supported by the IAEA was used to record the spectra acquired by an Inspector 2000 multi-channel analyser.

5.2 Data Analysis

An attempt to use peak areas by simply using the values displayed in the acquisition window of the IMCA software for a predefined Region of Interest (ROI) set around the 185.7 keV region of the spectra was attempted. This simple approach did not return acceptable answers as the high level of ambient background encountered at the facility significantly affected the values obtained from the simple peak interpretation used in the acquisition window. This is illustrated in Figure 5

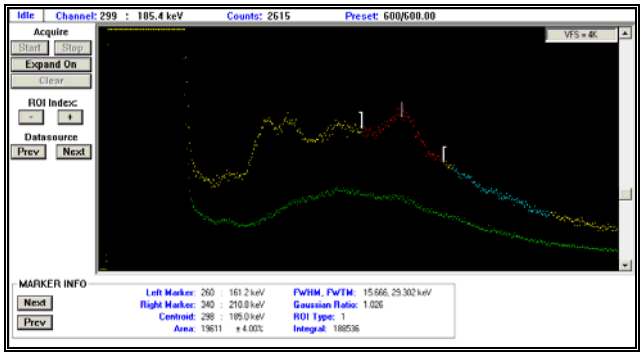

Figure 5: IMCA Acquisition Window and ROI Area information.

Consequently a full peak analysis based on the peak erosion method (GAMMA-M library) supported by the Genie 2000 software [7] was performed to derive peak area in the 185.7 keV region of the NaI(Tl) spectra. A similar analysis was hence performed on the benchmark spectra to rederive the benchmark factor of the measurement system based on that peak analysis (as shown in section 3.2). Table 4 summarizes the measurement results for 5 different types of pellets.

Pellet Type ID	Declared ^{235}U [%]	Calibration Coefficient [^{235}U-%].s^{-1}	Count Rate [s^{-1}]	Bkg. Rate [s^{-1}]	Net Rate [s^{-1}]	Measured Enrichment	Relative Difference
Type-1	0.711	**0.0305**	*33.000 ± 2.50%*	*9.76 ± 2.75%*	**23.356 ± 4.92%**	**0.713 ± 6.93%**	**0.32%**
	0.711	**0.0305**	*26.700 ± 2.75%*	*4.76 ± 5.39%*	**21.944 ± 3.54%**	**0.670 ± 6.03%**	**-5.74%**
	0.711	**0.0305**	*26.120 ± 2.81%*	*4.76 ± 5.39%*	**21.364 ± 3.64%**	**0.652 ± 6.09%**	**-8.23%**
Type-2	0.711	**0.0300**	*26.444 ± 1.68%*	*0.96 ± 20%*	**25.482 ± 1.90%**	**0.765 ± 5.24%**	**7.63%**
Type-3	0.711	**0.0234**	*34.889 ± 1.67%*	*0.96 ± 20%*	**33.927 ± 1.81%**	**0.795 ± 5.21%**	**11.85%**
Type-4	0.711	**0.0558**	*22.667 ± 3.41%*	*9.64 ± 8.29%*	**13.022 ± 8.55%**	**0.757 ± 9.84%**	**6.43%**
Type-5	0.711	**0.0581**	*14.556 ± 3.65%*	*0.96 ± 20%*	**13.593 ± 4.16%**	**0.759 ± 6.41%**	**6.74%**

Table 4 Summary Results of the Benchmark Measurements

Overall, the calibration coefficients derived by MCNP and benchmarked against some reference material returned good results for the limited sets of pellets used in this work. Measured enrichments consistent, within the calculated uncertainty, with the declared enrichments were derived for 5 type of pellet presenting different geometrical and physical characteristics. This proves of the ability of MCNP in accounting properly for the relatively small geometry changes between pellet types and hence of the advantage of numerical calibration not relying on reference materials absolutely identical to the measured unknown in order to perform reliable quantitative measurements.
Measurements performed in high ambient background where properly analysed by Genie 2000 to derive consistent peak count rates between pellet types.

6 Future Work

The experimental work undertaken at facilities showed the necessity to have a reliable spectrum analysis routine for deriving appropriate peak count rates. Within the framework of the analysis packages currently supported by the IAEA this would imply to upgrade some of the existing software to integrate background subtraction capability along with peak area analysis. This in itself does not present any technological challenge. The interpretation of a measured count rate by a selected enrichment calibration constant depending on the type of item being assayed could also be integrated into a seamless measurement/analysis package, which could open the possibility to implement numerical calibration on a much wider basis and for a large variety of items. This calibration method could therefore be extended to any geometrically well-characterised items even if presenting a broader range of enrichments.

7. Conclusion

The present work has described the process used to provide a reliable numerical calibration method for quantitative NaI(Tl) uranium enrichment measurements of uranium di-oxide pellets. This approach constitutes for the first time an effort among the IAEA to support the use of Monte Carlo techniques for standard inspection work. Such a development allows extending traditional calibration approaches, which require the use of fully representative reference material for calibration purpose, and hence could offer more flexibility in calibration capability. This in turn would result in direct quantitative estimates to be performed on more regular basis during inspection. Preliminary results from test measurements at fuel production facilities have shown the good performance of this method. Upon detailed review of the performance of this method further work will be performed to improve the reliability of the results and to streamline the use of enrichment analysis by inspectors so that the enrichment result could be made readily available from a simple to use analysis software.

8. References

[**1**] X5 Monte Carlo Team, “MCNP – A general Monte Carlo N-Particle transport code, version 5”, Los Alamos National Laboratory, LA-UR-03-1987, 24th April 2003.

[**2**] Drawing of 51 BS 12.7 / 2E3-XamT Sodium Iodine Detector, Reference V-0130-1 Scionix Holland B.V.

[**3**] Canberra Model 7419 Shield/Collimator (see http://www.canberra.com/products/523.asp)

[**4**] "ISOCS Calibration Software“, User’s manual version 4.0, 9231013D, Canberra Industries (2002)

[**5**] IMCA nuclear Safeguards Software, User’s manual, 9231182B, Canberra Industries (2001)

[**6**] "Genie 2000 Spectroscopy System“, Operations manual version 2.1, 9230846H, Canberra Industries (2002)

[**7**] "Genie 2000 Spectroscopy System“, Customization Tools, 9230847F, Canberra Industries (2001)

Session 12

Information treatment / security - remote monitoring

System Level Security Assessment

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Abstract:

Any application that runs on a network has the potential to act as a conduit into an entire remote monitoring system and therefore every application on that network must be designed and implemented in a way to maintain a system's overall security requirements. As unattended and remote monitoring applications are dependent on overall security to maintain the integrity of the safeguards data, knowing that each element is secure should be a critical step in the design and deployment of the system. This is especially true with systems sharing a network infrastructure with other environments or when a gateway onto the Internet is present. This paper will address the benefits of using security assessments within remote monitoring applications, identify where the potential vulnerabilities lie and the methods used to aid in the detection of these problems.

The Next Generation Surveillance System (NGSS) User Requirements Document 2.0 from March 2004 lists several requirements for an Unattended or Remote Monitoring System. This requirements document will be the basis for illustrating potential system vulnerabilities. This paper will discuss general vulnerabilities and provide methods the IAEA and others can use to verify systems compliance to the required security parameters.

Keywords: NGSS, security, remote monitoring, vulnerabilities

1.0 Introduction to Security Assessment:

The security assessment can illustrate potential weaknesses within the system ranging from application level problems, core operating system problems, and network deployment issues. A full security assessment is an integral phase of the installation of any network-based application and consists of:

- A vulnerability assessment
- Component research
- System penetration testing
- Testing of misuse cases

A vulnerability assessment is the assessment of a systems potential susceptibility to unintentional or intentional compromise of the systems overall integrity. The details of the components used to create the remote monitoring system are researched to understand the inner workings of each component. This research includes analysis of the operating systems used, the deployment configurations, and the applications running on each component. A system penetration test is a controlled simulation of an intruder attack and is used to assess the network or application susceptibility to external attack; additionally, it can be used to verify a system's compliance to the specified accreditations. Penetration tests can reveal that systems are not always running at the intended security level. For example, systems that are designed to run in modes like FIPS-140 (either level 2 or 3) will often be deployed without any security settings. A penetration test of the

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SAND # 2005-2902 C

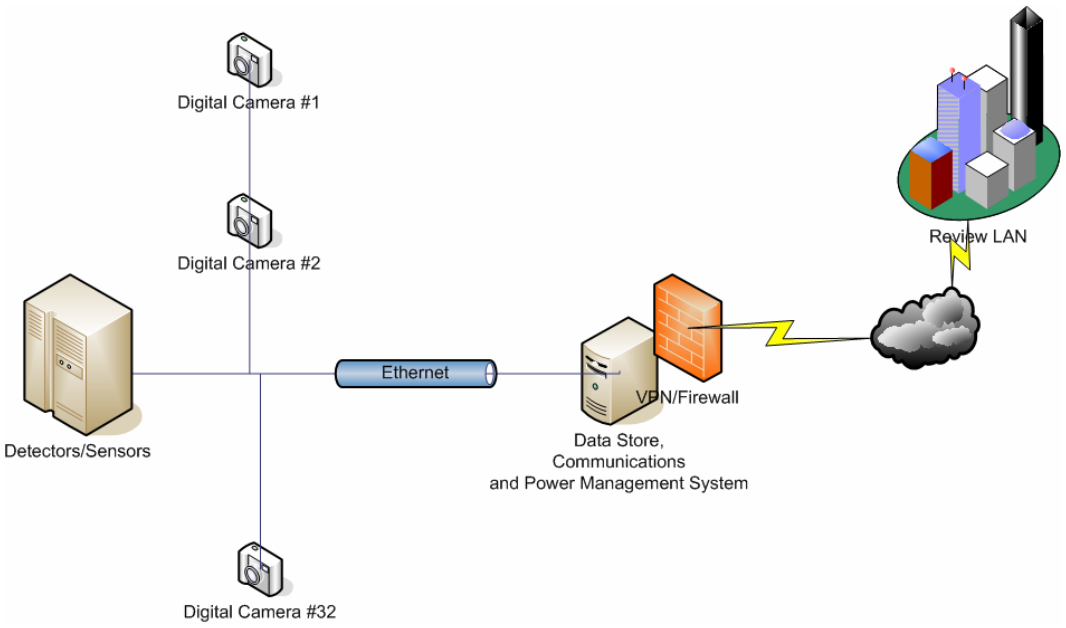
system as it is designed to be field deployed or even testing within the field is critical to verifying the levels of all components installed within the system. Testing misuse cases simulates a malicious attacker's actions against a target system, which can prove the existence of exploitable vulnerabilities that could lead to a system compromise.

A security assessment is used to identify all aspects of the network including all connecting end points and all networkable components. The assessment helps to ensure that each component secure in the event of a compromise of any of the other components. Also, the assessment verifies that each component and the interconnections between components is working at the highest level of security possible which will reduce the impact of any security penetration.

2.0 Definition of the Next Generation Surveillance System:

The Next Generations Surveillance System (NGSS) is a system designed to observe activities of nuclear material within nuclear facilities and to record these activities so outside inspectors can review the data and draw conclusions regarding a facility's compliance with safeguard treaties and protocols. The NGSS will be capable of operating in an unattended mode in which all the data is gathered and stored within the system and can be manually transferred without the use of a network to a review system. Additionally, the system will operate in a remote monitoring mode, which allows network transmission of the data to a remote review system.

In this paper, the NGSS will be simplified to comprise of an image-taking device (ITD), which will typically be a camera and a computer processor. Other components may include a radiation hardened camera, radiation tolerant camera, gamma cameras, a data consolidator, tamper indicating conduit, a data review station with software, tamper resistant and indicating equipment enclosures, as well as diagnostic monitoring software. Other sensors like balanced magnetic switches and heat sensors may also be used depending on the configuration. All components will communicate over Ethernet using TCP/IP. In remote monitoring mode, these components will be connected to a communications network to allow review of all data from the IAEA.



Proposed Remote Monitoring NGSS

2.1 System Specifics and User Requirements

The IAEA is concerned with the security of the data collected, stored, and transmitted within NGSS. The user requirement document specifies several security measures for the various components and data transmissions. These security measures include:

- The use of a public key infrastructure (PKI) based cryptography system using X.509 certificates
- The masking of data transmissions
- The use of standard algorithms
- The use of trusted compliers for security components

The document specifics that all security components must be FIPS-140 certified using PKCS #11 tokens. PKCS #11 tokens create an interface to the multiple cryptographic devices that are active within the system. This makes each cryptographic device look logically like every other device, regardless of the implementation technology. Thus each device can interact with other devices without the knowledge of how the cryptography has been implemented. This allows the simplest method for all devices to interact within the PKI system.

To protect the safeguards data against unauthorized access and tampering, the user document states that all images must be encrypted and all data transmissions must be masked to provide protection from an unauthorized person gaining knowledge of system settings such as the Picture Taking Interval (PTI). All transmissions schemes must be approved and are ideally encrypted and authenticated using standard algorithms. Additionally, any security component software must have the source code reviewed and compiled using only a trusted complier.

The review network can be located remotely or within a different network segment within the same facility. This network consists typically of a VPN/firewall which pairs with the VPN/firewall located within the NGSS network, and one or more personal computers. The nature of the review network has not been defined within the scope of the NGSS. Ideally this would be an isolated network. The software used for reviewing the data has also not been specified. The configuration of the VPN/firewall and the review system becomes critical, as it could potentially allow compromise of the entire system.

If a personal computer is used with in the system, the operating system must be approved and any running applications would ideally be limited to only those specifically required and approved. Embedded systems can utilize any operating system having the ability to interact with a standard Windows-based computer.

2.2 Potential Vulnerabilities Within a Generic NGSS Installation

To fully understand the level of security risk posed by an unattended or remote monitoring system based upon the proposed NGSS, it is vital to fully consider each portion of the system. The following sections provide a granular analysis of potential security weaknesses within the current generic environment and demonstrate the need for security assessments in the following areas: Software, Connectivity, Cryptographic Infrastructure, and Network Integration.

2.2.1 Software

Application issues

Within most applications the most prominent vulnerabilities are buffer overflows conditions and the ability to create a denial of service. A buffer overflow occurs when too much data is put into the buffer or temporary holding place for data within the computer's memory, which can result in a system crash or a back door leading to system access. A denial of service (DoS) attack is a type of network attack that attempts to render a network resource useless to users. Software which does not demonstrate these vulnerabilities using a verified complier, can do so when compiled using an unverified complier. Compilers handle libraries in different manners and if the complier has not been tested to verify these libraries, even well written software may exhibit potential overflow vulnerabilities once compiled.

General Personal Computer Operating System

Within the Microsoft Windows line of operating systems, security vulnerabilities have been discovered that include serious remote administration compromises and DoS conditions. If exploited by a malicious attacker, even a novice computer user with minimal levels of competence (I.e. a *script kiddie*) can obtain full control over this key network host. To mitigate these risks, it is vital that the administrator apply vendor security fixes and patches to mitigate these well-known security flaws. Ongoing vulnerability assessment will help to determine the patch level of such a system and identify emerging security vulnerabilities.

The various flavors of the UNIX operating system inherently hold a wide variety of issues depending on the deployment and configuration. These issues include remote compromise to the root level via buffer overflow attacks, denial of service conditions, and server application issues. Unix applications have often also been susceptible of localized elevation of privileges from limited access to that of the root level.

Components or appliances running embedded resources contain processors and memory much like their non-embedded counterparts. In addition, these devices are increasingly offered as networkable hosts, using operating systems and applications which have been customized for the component's specific purpose. Like any networked system, these embedded devices have been demonstrated to be vulnerable to common attack vectors and can be exploited and compromised depending upon configuration, functionality and the overall posture of the device within a network. Embedded systems are often prone to DoS attacks as well as remote or local compromises facilitating an attacker to take control of the device or escalate privilege from a limited user to root or admin levels; these conditions can be caused by buffer overflows, null pointer exceptions or general logic flaws. Some of the demonstrated exploits include the remote control of smart temperature sensors, unauthorized opening of a security door, and turning a printer into a covert network scanner.

Data Storage Module

Images and sensor readings from specialized network devices may be sent and processed by a data storage server. Storage methods may vary between implementations of remote monitoring system, including the potential use of a Relational Database Management System to gather bulk data readings. In general, this implementation will involve the installation and configuration of commercial database software, engaging potentially vulnerable network services, and thus providing additional penetration attack vectors. This reinforces the need for patching, maintenance, and secure configuration that is not only essential to be applied to the data server's base operating system but also to all network services that this host provides to the remote monitoring network. Specialized vulnerability software and a skilled security consultant can assess security weaknesses within a database system. Many database vulnerabilities and weak configuration practices can lead to full remote administrative compromise of this server.

Image Transfer Protocols

The process of transferring and reviewing stored sensor readings and images from the data storage server to users on the review network as well as the initial transfer of images from camera to server requires the additional network services on the data store module. Commonly, File Transfer Protocols (FTPs), network file sharing methods such Microsoft's NetBIOS, or web-based file, can achieve this. In each case, more network ports are opened providing the potential for more security vulnerabilities and opportunities for an attacker to gain control or affect the integrity of this host and the data contained within. There are many security issues with implementations of FTP that do not properly validate user input, allowing for DoS conditions or remote administrator compromise of this server. All known vulnerabilities can be discovered safely with good vulnerability assessment practices; new security holes can be revealed by a skilled security research team.

Unauthorized Access via the Power Management System

In the event a separate hardware based power management system is used to control the reliability of the data storage server should electrical power fail there are potential security issues management systems. Although in many cases this device may be hard wired to the server, it is also possible for this equipment to run on the remote monitoring network over TCP/IP. In this configuration, the internal operating system of the power management device may be susceptible to security vulnerabilities that facilitate network attacks in the same way a regular PC based host is. Compromise of the embedded operating system or unauthorized remote access to the device itself will allow serious disruption to the services the power management device provides. This will affect the integrity of the data storage module and the entire monitoring system environment.

Digital Network Cameras and Specialized Sensor Devices

Depending upon the model of camera used, various network attacks may be possible either targeted against or channeled through these devices. Digital cameras that are TCP/IP network compatible and transmit images via the network should be considered network hosts in their own right. Modern network cameras contain operating systems and network services for functionality and remote administration that can be vulnerable to attacks in the same way as servers. It may be possible to discover these flaws with automated vulnerability assessment tools; however, this is generally not effective due to the nature of the compact services running within embedded devices such as these. In general, vulnerability research and in-depth analysis can reveal security holes that will at the very least affect the reliability of network cameras and compromise the quality of image delivery. A worse case scenario would be to compromise vulnerable network services provided by the digital camera, leading to interactive access of the device's operating system and file system / memory. In this case it could be possible to launch network attacks against the monitoring system from the camera itself. This may circumvent network protection such as firewall / router access control or VPN trust relationships.

In exactly the same way it may be possible to attack or channel attacks through network-aware sensors that form the input peripherals to the monitoring system itself. An integral part of the monitoring system is the use of radiation detectors and specialist sensors. If these devices are developed to be network-aware, then processing the TCP/IP stack and utilizing self contained operating systems and memory (all normal network security considerations) apply. Due to the criticality of these devices, great care needs to be taken to ensure that vulnerabilities are not present within these hosts. Although using network enabled specialist sensors will greatly aid the administrative process, it may not be practical to manage the risk associated with compromise of these devices.

Network Infrastructure Devices
The monitoring network requires the use of network infrastructure devices such as routers, switches, and other managed devices. Each of these contains an embedded internal operating system and, unless properly configured and secured, this operating system is vulnerable to compromise. Compromise of these services can lead to unauthorized access and malicious configuration by attackers. The impact of this can vary greatly, depending upon the device capability; at the very least the integrity of network traffic delivery can be affected and subverted. Again, any known vulnerabilities can be discovered by the vulnerability assessment process.

2.2.2 Connectivity

The backbone of a remote or unattended monitoring system in the NGSS configuration is Ethernet based. In general, it can be assumed that physical access to a network is limited by the nature of the installation; however, this may not always be the case. Local network considerations should include the possibility of an attacker plugging directly into the infrastructure. Based upon this attack vector, several options and safety measures can be employed. It is possible to passively monitor a local network by *sniffing* network traffic and analyzing for salient information leakage. The use of cryptography can mitigate the risk of data compromise, but will not prevent basic network discovery, mapping, and general intelligence gathering. When switched network infrastructures are used instead of shared network mediums (hub connectivity), traffic analysis becomes more complicated. Analyzing network traffic within a switched environment may be a more involved process but is certainly achievable with advanced techniques. The use of a vulnerability assessment and penetration testing will aid in determining the existence or nature of any weakness with in the network backbone.

Certain implementations of a remote monitoring system may require special networking considerations. If a wireless link is ever deemed necessary to connect portions of the monitoring environment, the potential for network penetration is greatly increased. Wireless networks are notoriously susceptible to interception and insertion attacks that will allow for network penetration by a malicious attacker. Wireless security assessments can discover potential opportunities for remote compromise and close or remediate vulnerabilities. If wireless technology is ever deployed as part of an unattended or remote monitoring system, a wireless security assessment should be sought in addition to vulnerability assessment and penetration testing activities.

2.2.3 Cryptographic Infrastructure

Attacking the Trusted Certificate Authority
Digital cameras within the monitoring system store an encryption key from a trusted Certificate Authority (CA) in a secure fashion within their memory. The encryption key is a vital part of the data integrity and confidentiality process required for creating images within the monitoring system. It may be possible to attack the CA directly if it is part of the monitoring system or an interconnected network. This is entirely probable because the CA needs to be in communication with the digital camera peripherals and may even run as part of the data storage server. If the CA is compromised, a malicious attacker may be able to issue certificate revocation commands to certificate peers, in this case the digital cameras. Once a certificate has been revoked, a new (false) certificate may be issued, compromising the secure configuration of this procedure. If the CA cannot be attacked directly, it may be possible to falsify the CA's credentials and transmit artificial commands to the camera.

2.2.4 Network Integration

Circumventing VPN Protection
The security of Virtual Private Networks (VPN) lies within the configuration of a mutual security association between end-points. It may be possible for an attacker to affect this process to subvert the encryption and authentication provided by a VPN solution. Under the right circumstances it is feasible for a hacker to utilize a *Man-In-The-Middle* style attack. This involves the attacker inserting themselves transparently between the two endpoints during the security association process. The attacker essentially fakes the negotiation handshake by masquerading as each endpoint and negotiating the level of security down to an unencrypted level. This style of attack is complex in nature, but is a valid method of testing VPN security during vulnerability assessment and penetration testing. If the VPN can be circumvented in this way, the data traveling between the monitoring and review network will be visible to an attacker. If the cryptographic implementation is not secured to the specification required an unauthorized user might be able to reconstruct valuable sensor and camera data.

The Review Network's Insecurities
In the case of an unattended monitoring system, there is no direct link to a review network; the data is delivered *by hand* to the review network for analysis. However, this process has a large administrative overhead and has called for a move towards a remote system that directly connects the monitoring system into a parent or review network in a secure fashion. Consideration has to be given to the security of the review network as it facilitates a conduit into the remote monitoring system (and vice versa). The authentication of users that have access to the monitoring network has to be controlled to ensure that malicious internal attacks, either accidental or deliberate, can be traced and mitigated. Furthermore, if the review network has other gateways or access points into its environment, these also have to be assessed to determine the risk of an external attacker penetrating the review network, leveraging access to the monitoring network and compromising its integral business functionality. This may seem to be an extremely complex process but is in fact trivial to exploit with only minor security configuration issues. The process of external penetration testing will assess the susceptibility of this form of attack.

Inspector's Laptop
In the event that the IAEA inspector will be using an Agency-provided laptop for data collection, review, and configuration of the NGSS, this system, if not protected, could provide unauthorized access. If this computer, which will be running a windows operating system, has contracted a Trojan or a virus, these could be transmitted onto the NGSS during data transmission or configuration. Also, if the laptop is configured to allow split access between two networks, it can become a router to allow access from an insecure network onto the NGSS network. Via this access, an unauthorized person could potentially gain access to the data or alter settings within the NGSS.

3.0 Methodologies:

The potential vulnerabilities that have been discussed so far can be discovered and tested for by using vulnerability assessment, penetration testing, and specialist auditing techniques. There are many different aspects required to produce a thorough security assessment and many tools/utilities that can help. This section provides a brief summary of the activities involved to secure the installation of a computer network. Specialised security auditing can assess the potential for compromise of the proposed remote or unattended monitoring system, resulting in clear and concise recommendations to remediate security and design flaws.

3.1 Host Based Vulnerability Assessment

Each individual network host, regardless of function, is subjected to an in-depth systems analysis. The methods or type of input and output to each device is assessed and as are any network services provided. In the first instance, the assessment would establish the type and function of the host before enumerating further critical data to build possible attack vectors. Weak configurations are checked to ensure that an attacker cannot access a host via something as simple as a blank password or a well-known manufacturer default back door. Vulnerability checking software is utilised to discover exploitable security holes that are known within the public domain. If no vulnerability has been discovered for a particular service offered by a network host, a talented security consultant might be able to establish a new flaw. This is generally achieved by provoking deliberate misuse cases, specifying spurious or unsafe input, and debugging the observed outputs from the service. If vulnerabilities are discovered, publicly known or fresh (zero-day), then exploit code can be developed to lever access or gain full control of a vulnerable host. This entire assessment process is focused upon producing detailed steps to fix security weaknesses.

3.2 Network Traffic Analysis

In addition to vulnerabilities that exist within network host, it is possible to gain valuable data by monitoring the network traffic flowing through the Local Area Network (LAN). Many standard network protocols are unencrypted or said to pass *in the clear*. This can reveal host names, address details, configuration options, and even system passwords. Software to *sniff* and analyse network traffic is freely and readily available. A security assessment would include the operation of network equipment in *promiscuous* mode to receive all network traffic, even that which was not original destine for the monitoring system. This also verifies that each device has met the requirements of not allowing track able transmission patterns. Additional methods will be used for any additional network mediums, like wireless systems.

3.3 Network Access Control Auditing

Networks that utilize firewalling, simple Access Control Lists, or trust relationships need to have these elements audited. This process can be extremely complicated and generally requires a specialist in this field of assessment. This assessment would reveal where a perceived configuration and actual configuration may differ resulting in a perimeter security hole.

3.4 Remote Penetration Testing

A remote penetration test simulates a blind attack from a public domain (usually the Internet) with very little or no prior knowledge of the system. It is considered to be one of the most difficult assessment techniques, but it also offers the highest assurance that complete network security has been achieved. Specialist security consultants will gather a profile of the target from the pubic domain and launch a series of probes looking for security weaknesses to gain access to the target system. Penetration testing simulates as accurately as possible the susceptibility to a hostile attack from an external force.

3.5 Trusted Misuse Case Analysis

Common perceptions of the network security threat focuses solely upon the risk of attack from outside of the facility. A large percentage of problems or misuse attempts come from within the facility or network itself for a variety of reasons. Whether accidental or deliberate, the consequences of attack from internal misuse can be even more effective than those posed by external influences. Trust relationships or access control will generally favor internal network users and not provide any protection from attacks that originate from this *trusted* realm. In addition, if a system employs monitoring techniques such as the use of Intrusion *Detection Sensors* it is less likely that administration staff will monitor trusted users and instead monitor only for an external threat. To assess this area of concern, a security consultant will work closely with the system designers and maintainers to establish a series of misuse cases and then reconstruct these to test susceptibility to these malicious scenarios.

3.6 System Specification Policy Review

A further security assessment technique is a policy or specification review. This checks the adherence to original system specifications and audits secure operating policies. Any deviance from the prototype definition can be cataloged and either justified or rectified during remediation phases. If a lack of security policy or secure operating methodologies is apparent, the security consultant can again work in partnership with system designers and maintainers to develop these working practices. This is critical to insure that each system is meeting the user requirements for the NGSS.

The final test of any remote monitoring system would be to conduct a network attack in situ. Parties outside the development team with the goal to defeat the system would conduct this attack. This test would not only verify the systems deployment, but also assess all the operational procedures, the system's life cycle and maintenance. If a remote monitoring system has been well designed and deployed meeting all the user requirements, this attack should be unsuccessful.

4.0 Conclusion

Security assessments, especially when preformed by an outside auditor, are critical to having a secure system. The security assessment will not only determine any potential problems which can be resolved before deployment, but can also verify a developed system has met the minimum requirements. A complete requirements policy document should not only include the development requirements, but also the testing or verification, the deployment, retesting, and continuous monitoring of the system. The retesting will determine if any changes or new potential problems have occurred which is highly critical within a system that can have devices added or removed automatically. Continuous monitoring of a system once it has been tested provides the knowledge that the systems security state has not changed. It is synonymous with the physical tamper indicative housing. Without including a security assessment as part of the entire system plan, there exists the potential of serious malicious misuse.

5.0 References

Attacking Network Embedded Systems, FX & KIMO Phenoelit, Blackhat Breifings, Las Vegas 2002 http://Ciac.org
Embedded System Security, Philip Koopman, Carnegie Mellon University, Embedded Computing July 2004
Next Generation Surveillance System, Version 2.0, Division of Technical Support Department of Safeguards, IAEA March 8, 2004
Security Code Review Guidelines, Adam Shostack, August 1996 republished May 2000.
http://www.NGSSoftware.com
http://www.securityforest.com/wiki/index.php/Category:Source_Code_Scanners

Implementation of Remote Monitoring in Canada

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Abstract:

Nuclear Generating Stations (NGSs) in Canada are safeguarded using bundle counters, core discharge monitors and surveillance equipment. Until recently, data from these systems was not available off-site. The technology for remotely accessing the data (often referred to as remote monitoring (RM)) has been available for a long time, but there have been problems finding solutions which are cost-effective and available at the sites in question.

A number of studies have been made to determine the most cost-effective means of transmitting the data from the stations. The use of traditional communications services can be expensive for the relatively low data volumes being considered; they may also suffer from obsolescence in the near future as the telecommunications industry moves increasingly to internet based services. Local access methods such as ADSL (Asymmetric Digital Subscriber Line) are now widely available in populated areas, but these consumer oriented services are not generally available at the NGS locations. A further option is to find a way of sharing the communications infrastructure operated by the NGS.

Security issues need to be addressed in implementing an RM solution. The IAEA is mainly concerned about the data it collects being authentic, whereas the state being monitored may be more concerned about the confidentiality of information being passed over the internet, particularly when it contains image data. In addition, in the situation where the data passes over the facility's network, the facility needs to make sure that its own interests are protected.

The paper provides an overview of recent and planned remote monitoring implementations in Canada. Two general approaches are described; one using the existing telephone wiring and the other involving sharing the facility's network.

Keywords: safeguards**;** remote monitoring; unattended monitoring systems; surveillance; ADSL; VPN

1. Introduction

Canadian Nuclear Generating Stations employ CANDU on-load reactors; reactors of this design routinely load fuel and discharge fuel every day. To monitor movements of discharged fuel, the IAEA uses Core Discharge Monitors (CDMs) and Bundle Counters, functions which are carried out by VIFM equipment. Movements of used fuel are further monitored by surveillance cameras at strategic points as it moves through wet storage and is ultimately transferred to dry storage. By the end of 2005, all surveillance will be carried out using the DMOS and DCM14 cameras. In the past, data from both systems has been collected by IAEA inspectors every three months during inspections. The data collection activity is not only time consuming for the inspector, but intrusive for the facility which needs to provide escorts. Also, due to the minimum delay of three months before reviewing data, equipment problems may not be noticed until it is too late. The ability to remotely access data from safeguards equipment has been a long standing goal for both the IAEA and the Canadian State System of Accounting and Control (SSAC). The reduction of inspector effort on technical activities will result in efficiencies for both the IAEA and the facility. Access to current data from a facility and the Status of

Health (SOH) of equipment on a daily basis will give the IAEA more timely and more reliable information.

Both DMOS and VIFM systems have been accepted by the IAEA for operation in RM mode. Although there are a number of well established solutions to make the necessary local access connection, they may not be practical or cost-effective. It has been found that some flexibility is needed in choosing the most cost-effective method to close the last few kilometres of the connection at the facility end.

2. Remote Monitoring Requirements

Canadian NGSs are located primarily in Ontario but also in Quebec and New Brunswick. Canada has a highly developed telecommunication infrastructure, but some otherwise suitable technologies may not be practical at the NGS locations.

2.1. Canadian Facilities

In Ontario, Canada has three multi-unit stations, Pickering (8 units), Darlington (4 units) and Bruce (8 units). In the case of Pickering and Bruce, each unit has an associated VIFM system; as a result, the VIFM cabinets are distributed within the facility. Currently, there are also two dry storage facilities, Pickering and Western Dry Storage (located at the Bruce site), each has a single DMOS.

Single unit stations of the CANDU-6 design are located at Gentilly in Quebec and Point Lepreau in New Brunswick. Each station has a VIFM system and a DMOS which are located in the same room.

Apart from the Pickering NGS, all NGSs are located well away from major urban development. Pickering is close enough to the IAEA's Toronto Regional Office that is within local calling distance and is toll free.

2.2. Data collected

Currently all operational VIFM and DMOS systems installed in Canada report Status of Health (SOH) daily. Since the data volume is small, this can be carried out by a modem connection over a conventional telephone line. The connection is secured by a VPN connection established by the IAEA. The SOH messages provide a daily re-assurance that the equipment is working and collecting data; no other data is transferred.

The data volumes collected by the VIFM and DMOS systems depend on their exact configuration. VIFM systems can contain either or both bundle counter and CDM functions; not only do the numbers of detectors vary but so do the gate times used for particular installations. For planning purposes, the daily data volume for a VIFM system is estimated at 5 Mbyte/day/reactor. This includes summary data (list of transfers or discharges), raw data (detector counts) and SOH information as well as the overhead required to operate the VPN.

The volume of data collected by a DMOS is dependant on the number of cameras and the picture taking intervals assigned to those cameras. For planning purposes, we have estimated 15 Mbyte/day/camera. Based on these assumptions, DMOS systems can be expected to generate a data volume in the range of 100-200 Mbyte/day. Even with the toll free lines, which are available in some instances, the time taken to download such data volume over a dial-up modem becomes a problem (in excess of 10 hours). Since the IAEA plans to complete the downloads for all Canadian sites overnight, perhaps between midnight and 6 AM, the scheduling problems become evident. The need for faster downloads is not only a cost-efficiency issue, it can be an operational necessity. To move data from the Toronto Regional Office to the IAEA Headquarters in Vienna without impeding other business and achieving highest transfer rates, the Agency will need to do so during low traffic periods and it will need to complete the transfers quickly.

2.3. File Structure and Security

Earlier versions of the VIFM collect software (Version 4.6 and before) were based on a fixed file size for raw data in order to optimize efficiency and reliability. Using these systems, data recorded over a 24 hour period need not fill a file or may fill multiple files. The use of such a file system when data is being downloaded on a daily basis could lead to problems when the data is reviewed at the Agency. The file system incorporated in the latest version of the VIFM software (5.1) is designed around the IAEA's need to download files on a daily basis. SOH reporting is also now compliant with the IAEA's standard format. This will allow the IAEA to used automated means to assess the SOH data and trigger alarms when there are equipment problems.

The IAEA is responsible for authenticating and encrypting data before it leaves the IAEA equipment cabinet. The IAEA has adopted standardized approaches based on Cryptographic Message Syntax (CMS) and S-MIME for doing this. The VPN within which this data flows provides additional security.

3. Data transmission options

3.1 Traditional data transmission

A variety of methods exist which could be applied. Some would not be cost-effective due to the relatively low volume of traffic. Detailed studies have shown that solutions involving the use of ISDN would have been quite cost-effective for the type of traffic that the IAEA would want to send. Unfortunately, the IAEA has not had good experiences with ISDN, there seems to be some inconsistencies on its implementation. With the move in the industry to internet based services, there is also a concern that ISDN style services may not be as well supported in the future.

3.2 Securing the path

The IAEA generally implements Remote Monitoring by creating a Virtual Private Network (VPN) over the internet; this provides a secure a channel for their communications. Netscreen VPN boxes are configured to operate only between the fixed IP (Internet Protocol) addresses assigned to the monitoring equipment and the server located at the regional office. The need for fixed IP addresses can normally be accommodated at a cost; basic internet services normally assign IP addresses when the connection is made. Note that the data travelling within the channel established by the VPN is also authenticated and encrypted at its source prior to transfer to the Netscreen VPN box. This part is routine procedure for the IAEA; however, flexibility is required in the method of making the physical connection with the Netscreen installed in the facility, the "last mile".

3.3 'Last Mile" solutions

A variety of "last mile" technologies have become available in recent years. Most people are familiar with ADSL, one of the family of DSL (Digital Subscriber Line) technologies. With this technology, many residential and small business users can have at their disposal the ability to transfer data at substantially higher rates than is possible with a standard telephone modem. The technology has some constraints; it has to run over copper wire and the length and quality of that copper wire limits performance and ultimately the availability of the service. The availability of the service is also subject to laws of economics; the provider needs a certain level of demand before it is cost-effective to support the necessary infrastructure. Note also that the ADSL service is asymmetric; maximum download rates are higher than the upload rates, unfortunately the latter being especially required for Remote Monitoring. At the telephone company end of the copper wire, a DSL Multiplexer (or DSLM) is used to decode the signals and pass them onto the internet. The NGS operator will have extensive telecommunications services both for voice and e-mail. These needs are often being met by fibre optic cables rather than traditional wire based means and cannot carry the analogue signals required for DSL.

An interesting wireless alternative was available at one of the Canadian sites. A locally operated Internet Service Provider (ISP) was able to provide an 802.16 (known as Wi-Max) service. Its cost and performance was similar to that offered for commercial ADSL service. In this case, the NGS had already used this service to supplement their existing network for special operations involving

transmission of large data sets. According to the ISP there should have been no problems in transmitting the protocols required for a VPN over this type of link.

The “last mile” problem may also be solved by sharing the network facilities that the NGS operates for its own purposes. Since the network is operated to serve the operator, there is no obligation for the operator to agree to such an arrangement.

4. Typical Implementations

RM is in the process of being fully implemented at Canadian NGSs. At some sites it is fully operational; the others should be operational before the end of 2005. The IAEA’s standard approach is to use a VPN applied over the internet, however, it has been found necessary to resort to various means to make the final connection to the IAEA equipment. Three types of implementation have been encountered so far.

4.1 Standard DSL

This would appear to be the ideal option for the IAEA. It supplies an adequate bandwidth at a reasonable cost. It is also completely independent of the facility’s network infrastructure. As explained earlier, it depends on a copper wire connection to reach the telephone company’s exchange (or Central Office) where it is connected to a DSLM. From there it is transferred to an internet service provider, possibly the telephone company itself. Only one of the sites in question has any remaining copper based services and these were free because traffic had been transferred to other channels. Also in this case, the wiring to the telephone company’s central office was short enough to support the DSL service. See Figure 1. Note that there are two runs of telephone wiring, one inside the facility, and the other from the facility to the telephone company’s Central Office.

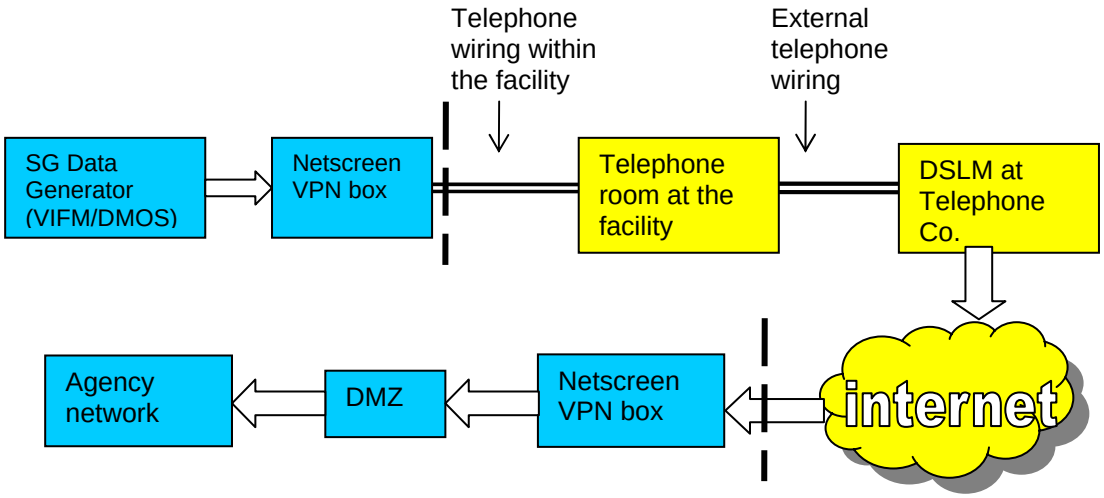

Figure 1 A conventional ADSL implementation

4.2 DSL On-site

The implementation shown in Figure 1 is not possible at most of the facilities due to the lack of adequate wiring reaching back to the telephone company’s Central Office. The connection from the facility to the Central Office is usually a high capacity trunk such as a T-3. A solution that has been implemented at a Canadian facility is to install the DSLM within the facility, see Figure 2 below. At this point within the facility, the telecommunications provider now becomes an internet service provider. For the user, the service remains the same as the first case. In fact it should be cheaper since he no longer has to pay monthly charges on a phone line in addition to the monthly internet service provider cost. Both the facility operator and the Agency retain totally independent channels. The cost of installing the DSLM may be an issue since it is an expensive piece of equipment which is designed to service (for example) 24 separate ADSL connections. There be other customers (such as contractors

to the operator) for these services which will make it commercially viable for the internet service provider.

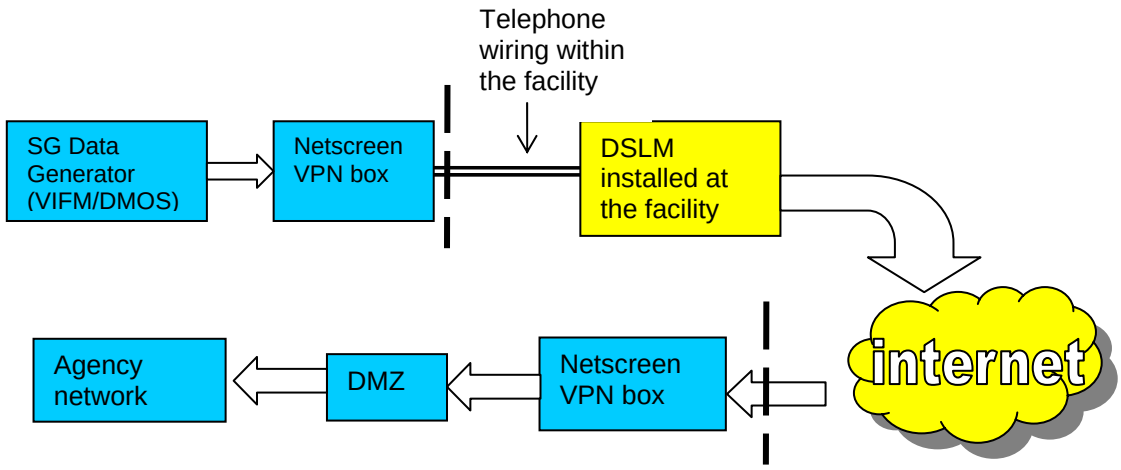

Figure 2 Modified ADSL implementation, the DSLM is installed at the facility

This solution requires no new wiring and will easily accommodate further ADSL connections. It is an attractive option which may find further application at Canadian facilities.

4.3 LAN/WAN Sharing

The sharing of the facility's Local Area and Wide Area Network (LAN/WAN) may be acceptable to some operators. Figure 3 shows this arrangement; note that the telephone wiring is redundant. There are of course some serious concerns on the operator side; they have to be sure that their infrastructure is protected. This can be done by confining traffic in various ways. Firstly, the IAEA traffic is restricted to specific parts of the facility's internal network by means of a Virtual Local Area Network (VLAN). At the facility's firewall only the traffic associated with IAEA's VPN is permitted to pass, primarily IPSec packets. The firewall further restricts the traffic to flow only between the designated points, the fixed IP addresses assigned to the equipment and the IAEA server.

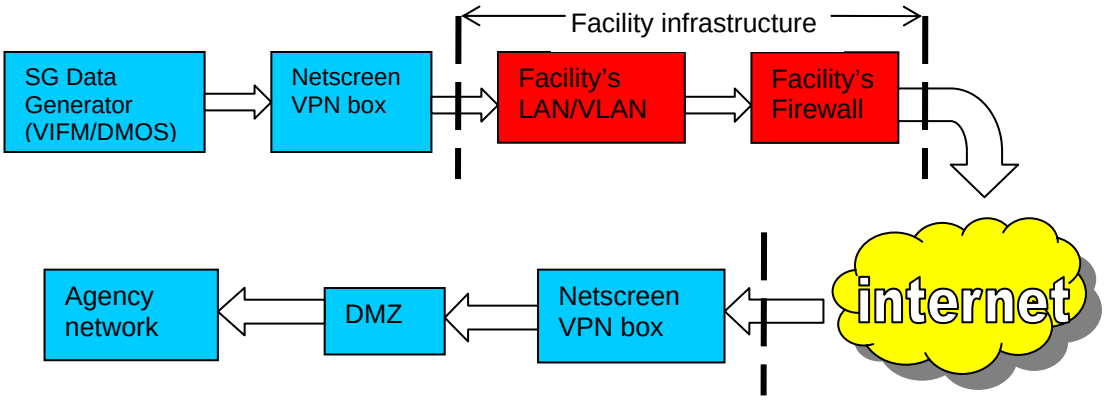

Figure 3 – Sharing the facility's LAN/WAN infrastructure

This solution is particularly attractive where the IAEA cabinets are located near to existing LAN services or where suitable extensions can be easily made. It should noted that minor interruptions in the facility's network operation should not be a serious concern to the Agency since data is always retained on the collect systems and any missing data will be forwarded to the Agency once the connection is restored.

5. Conclusions

Data services normally available in Canada may be difficult to obtain commercially at the facility since they are located away from major population centres. Two major alternatives have been identified. Starting the DSL service at the facility rather than the telephone company's offsite Central Office is an attractive option for all parties. It allows both the IAEA and the facility operator to have totally independent arrangements. Sharing the facility's network is an attractive option where there are a number of locations from which data has to be collected and those locations can be easily reached from the existing network wiring. For this to be acceptable to the operator, special arrangements need to be made to ensure that the security of facility's network is not threatened in any way.

Using the options described in this paper Remote Monitoring should be fully implemented at all Canadian NGS before the end of 2005.

6. Acknowledgements

The solutions reported on in this paper are the collective effort of technical experts employed at Bruce Power, Ontario Power Generation, New Brunswick Power, Hydro Quebec and the IAEA. The Canadian Safeguards Support Program would like to thank these individuals for their respective contributions.

Scope for Remote Monitoring in Large Scale Plutonium Facilities

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Abstract:
The sophisticated, integrated safeguards instrumentation and surveillance systems that are a key part of the multi-layered safeguards arrangements in large scale plutonium handling facilities such as reprocessing and MOX fuel fabrication plants have been described in previous papers/presentations. Such 'safeguards in depth' consumes large amounts of DGTREN inspection resources in on-site activities relating to the collection, review and assessment of data from the equipment involved. The objective of pursuing the remote transmission of safeguards related data in these circumstances is therefore to:

- *Make it easier for the safeguards inspectorate to collect and review data from safeguards systems and equipment, without compromising the confidentiality or the integrity of the data;*
- *Improve the utilisation of the safeguards systems and equipment involved (e.g. by enabling rapid reaction to equipment failure and/or preventative maintenance);*
- *Reduce the impact on operator resources and plant operations by less inspector physical access to plant;*
- *Reduce the radiation dose for inspectors and accompanying operators/escorts.*

Thus remote monitoring offers the potential to improve the effectiveness and efficiency of on-site inspection activities, both from an operator and an inspector perspective (a 'win-win' situation).

Keywords: encryption, safeguards; data transmission; remote monitoring.

1. Introduction

The first step in developing such remote monitoring installations at Sellafield was to enable the transmission of equipment 'state of health' signals from the UK to the DGTREN safeguards inspectorate in Luxembourg. Success at this stage provided a sound basis for the transmission of more useful but potentially more sensitive information. This paper describes:

- The Pilot Study and how the data sets for transmission have been extended;
- What the safeguards inspectorate does to ensure the integrity of the data that are transmitted;
- How the resulting systems have been accredited (e.g. in terms of encryption arrangements/procedures) to satisfy national requirements for the protection of sensitive safeguards information;
- The expected development of RM and estimates of the benefits with respect to inspection and operator time that may be realised.

The paper will also explore how RM can make a contribution to the continuing evolution of safeguards approaches at installations in the UK.

2. Pilot Study

Both DGTREN and BNFL examined their activities relating to inspections and drew up a list of potential opportunities where some benefit could be realised. In general, it was found that investment would have to be made in order to be able transmit the information to Luxembourg, or there were perceived classification issues. However in the Sellafield MOX Plant (SMP) at Sellafield in Cumbria, DGTREN had already invested in an Ethernet network to link its extensive array of in-plant safeguards equipment to the safeguards inspectors' offices at the Sellafield site . SMP was therefore chosen as a test bed to implement a system of remote monitoring.

3. State of Health

DGTREN has considerable experience [1] in the use of state of health messages. These have been

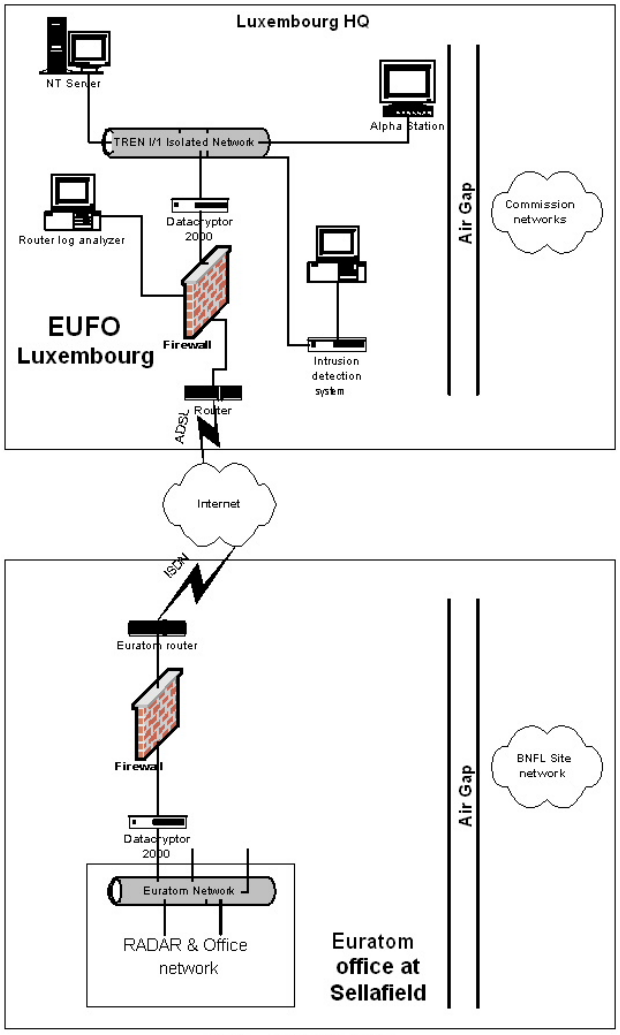

Fig. 1 - Overview of VPN

simple text messages from ASCII files transmitted from the installed digital surveillance systems in the plants and connected to telephone lines either standard or ISDN. A total of 47 video units are currently connected to dedicated PCs at headquarters. The information transmitted is in the form of simple messages related to the image taking performance and self-diagnostic errors. These messages have proved to be extremely useful in monitoring the performance of the equipment in the field. Based upon the information received DGTREN have been able to plan maintenance visits on short notice. This has 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.

Therefore, for the Pilot Study at Sellafield (see Figure 1) the first type of information that was considered was the 'state of health' data associated with DGTREN instruments. This indicates if, for example, an alarm, a warning or an error exists within a DGTREN Data Acquisition Module and consists of log and alarm files. The benefit is that DGTREN can take timely action to rectify faults, avoiding or minimising the requirement to, say, re-verify material. The information that is being transmitted is not classified and the DGTREN network is 'air gapped' at both ends from the BNFL and the DGTREN internal networks resulting in a minimal security risk.

The Pilot Study was very successful and it was decided to extend it to cover more sensitive information. This would have 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. Data Encryption

Extending the pilot study involved the electronic transmission from the facility and the UK of sensitive safeguards data, some which would include information subject to UK security classification. This meant that an initial issue was to identify a means of protecting transmission (e.g. appropriate cryptography), which would meet national requirements for the protection of such information.

The approach eventually adopted was based on the recommendations of the UK Government's National Technical Authority for Information Assurance, which recommended the use of the Datacryptor™ 2000. The Datacryptor™ 2000 product is specifically designed to provide secure communications over circuits at speeds up to 2Mbps using a variety of line interfaces. It is a synchronous line encryptor and prevents unauthorised information access and protects against eavesdropping for data transmissions using both private and public networks. The unit provides both tamper evidence and tamper resistance. If the tamper alarm is triggered, then the keys and the encryption algorithm are automatically erased. Once commissioned, the unit will operate automatically without further intervention. The product employs a Secure Key Management

scheme to securely generate and distribute data encryption keys. This dispenses with the previously time consuming tasks associated with secure key management.

5. Integrity of Data

The electronic data collected by the safeguards authorities on site at Sellafield is on a separate local area network from the main plant operating system. All data collected that is required for evaluation and assessment purposes will be passed through the dedicated telephone network link connecting the site to a Commission office in DGTREN in Luxembourg. The Luxembourg Datacryptor™ 2000 router will only allow communications between DGTREN headquarters in Luxembourg and the accredited DGTREN UK based isolated network. The integrity of data that will be received will be checked using currently available software that is compatible with the Datacryptor™ 2000 router.

6. Accreditation

One of the outcomes of the Pilot Study was to use the same process of UK security accreditation for the DGTREN transmission system both in the UK and Luxembourg as is applied for the BNFL network. It is UK government policy that all computer systems that process, store and transmit sensitive information are to be accredited against a set of standards laid down by the UK National Technical Authority. For the civil nuclear industry in the UK the Accreditation Authority is the Office for Civil Nuclear Security (OCNS), the industry's security regulator. The core of the accreditation process, which is risk management oriented, is the Accreditation Document Set (ADS). The ADS details the architecture, topology, processes, procedures, controls, protective security, and technical security measures applied to the system to give assurance that the risk to information within the system has been fully assessed and countered.

The ADS was submitted to OCNS and interim approval to operate the system was given in September 2004. Full accreditation has yet to be granted.

7. Learning Points

A number of lessons have been learned during this development of a system appropriate for the transmission of sensitive safeguards data from Sellafield to the DGTREN safeguards inspectorate in Luxembourg.
These include:

- identifying an encryption method which met national requirements - several methods were considered and rejected as unsuitable, wasting time and effort. The National Technical Authority was approached and a method appropriate for this application was recommended;

- translation of standards - it was necessary to ensure that the equivalent procedures/standards used within the Commission were referenced in the ADS. A GAP analysis was performed which confirmed that these were equivalent to those in place in the BNFL system. The risk analysis and calculations were performed by the BNFL IT Security Assessor;

- change of personnel - inevitably, during the course of a project that lasts several years, personnel will change. The ADS helped to capture relevant information and focus resources.

8. Future Developments

The original Pilot Study utilised the DGTREN systems being installed in the Sellafield MOX Plant. However, as the safeguards equipment in other large bulk handling/storage plants is upgraded the opportunities for implementing RM have increased.

Moreover, as safeguards approaches continue to evolve and improve (along with the associated technology), it is increasingly feasible to realise the potentially significant benefits of RM for both the safeguards inspectorate and the facility operator. Therefore, the application of RM in respect of installations at Sellafield which are inspected jointly by the DGTREN safeguards inspectorate and the IAEA is now also the subject of discussion.

The Sellafield nuclear site in Cumbria has a large number of material balance areas, many of which include item stores such as spent fuel ponds and product stores. The site also includes sophisticated safeguards measures implemented in plants, dealing with direct use materials, such as plutonium oxide stores and MOX fuel fabrication plants. These areas are of high sensitivity and physical access tends to be time consuming and requires stringent

security measures including inspectors always being accompanied by trained escorts.
The safeguards investments in these plants has been significant. For example in the Sellafield MOX plant the installed systems include large numbers of cameras, seals, and branches from operator systems such as identity readers, motion detectors, weighscales etc. NDA equipment is installed in line with the flow of the plant and includes neutron monitors and more sophisticated detectors for plutonium in powder, pellet, rod and fuel element form.

Another example is the fully automated Thorp plutonium store which has multiple systems with built in redundancy that have been operational since the mid nineties and include:

- verification of powder receipts by can weighing, non-destructive assay (NDA), identity checking and independent plutonium powder analysis in the safeguards on site laboratory;
- store penetrations are covered by a combination of seals, movement detectors and neutron monitors to detect unauthorised removal of PuO2;
- optical surveillance devices; can identification using bar code readers; eddy current reading of the sophisticated triple can package;
- inventory verification of cans in-situ in the storage channels;
- branching from the can contents (NDA) monitor for independent authentication.

These systems benefit the operator by minimising the impact of physical inventory verification. The Thorp store holds many thousands of plutonium cans and even a small sample size cannot be retrieved, verified and returned on a realistic timescale.

The level of inspection effort these safeguards measures require is significant. Typically DGTREN expends some 1000 inspection person days of effort at Sellafield each year. This number excludes the travel days which would add at least a further 40% to the person days total, given that travel takes place on Monday and Friday and inspection activities on Tuesday through Thursday. Remote monitoring offers scope to reduce this travel overhead and allow a greater degree of flexibility about when and how the information is reviewed. Further potential efficiencies would also be achieved by reducing the time overhead of accessing plant either for verification or equipment interrogation.

For the operator, remote monitoring offers a better risk management of potential inspectorate equipment failures and consequent re-verification burden. Reduced access also brings efficiency in reduced demand for escorts, support and plant personnel presence and associated radiation monitoring overheads. Once a secure system of remote monitoring is in place its use can be broadened to take any sensitive safeguards information traffic which has been approved by the state and the operator. This could include BTC's, reports, plant diagrams and drawings as well as routine safeguards reports and advance notifications

9. Conclusions

Remote Data Transmission of safeguards relevant data is proving to be a valid means of improving the efficiency of safeguards inspections without prejudicing their effectiveness. There are also other, fringe, benefits for both the operator and the safeguards authorities, which can be realised. These include:

- reduced risk of loss of assurance due to equipment failure by utilising state of health signal transmission;
- ability to perform remote software and firmware updates remotely;
- reduced burden on plant personnel required to accompany inspectors;
- availability of a secure route for transmission of other safeguards data (ICRs BTCs etc.).

9. References

[1] Kloeckner, Meylemans and Chare: Safeguards: Containment/Surveillance of Nuclear Material, the prospects of Remote Monitoring, 1st Annual European Energy and Transport Conference, Barcelona, 18-19 October 2001

Session 13

Facility specific Safeguards

Measurements of Low-Enriched Uranium Holdup at the Ulba Metallurgical Plant in Kazakhstan

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Abstract:

A recent joint effort of scientists from Los Alamos National Laboratory (LANL), the International Atomic Energy Agency (IAEA), and the Ulba Metallurgical Plant (UMP) determined uranium holdup at the fuel fabrication facility at UMP in Ust-Kamenogorsk, Kazakhstan. Measurements were taken with both high-resolution and low-resolution gamma-ray spectrometry systems. The focus of this report is the measurements made on the ventilation and vacuum systems. Low-enriched uranium (~ 3.3% in ^{235}U) oxide feeds the pellet manufacturing process at Ulba. Deposits are infinitely thick to the 186 keV gamma ray in many locations. This requires measurements of both the 186 and 1001 keV gamma-rays. The measurements used short count times, portable detectors, and portable MCA's with a wide dynamic range. The main goals of this joint effort were to measure as much of the ventilation system as possible, and demonstrate methods for measurement and analysis of holdup using low-resolution detectors and the Generalized Geometry Holdup (GGH) techniques. The current GGH approach is applied elsewhere for holdup measurements of plutonium and high-enriched uranium. The measurements at UMP are the first for low-enriched uranium holdup. This report discusses the measurement methodology, calibration of the measurement equipment, measurement control, analysis of the data, and the global and local assay results including random and systematic uncertainties. It also examines the differences in assay results between the low-resolution system using the GGH method and the high-resolution system utilizing the commercially available ISOCS analysis method. This project is funded by the United States Department of Energy Office of International Safeguards (NA-243).

Keywords: holdup; spectrometry; uranium; low-enriched;

1. Introduction

1.1. The Ulba Metallurgical Plant (UMP or Ulba)

A major part of the UMP [1] is devoted to the manufacture of low-enriched uranium (LEU) ceramic fuel-pellets for use in Russian designed nuclear power reactors. Ulba offers a variety of uranium material processing services as well. The fuel-pellets produced at UMP vary in enrichment from 1.0% to 4.45%. Currently UMP makes fuel pellets for the RBMK and VVER reactors. The plant uses various feed materials, including UF_2, U-oxides, and UNH, to produce its fuel-pellets. The plant also produces ceramic grade UO_2 powder, which it sells to numerous countries. The International Atomic Energy Agency (IAEA) inspects UMP for compliance with the Nuclear Nonproliferation Treaty.

The focus of this document is to report on the measurement of uranium oxide holdup in the process equipment, ventilation system, and vacuum lines of the plant made during two campaigns at Ulba in 2004, the first in July and the second in October. Holdup is defined as the material that adheres to the walls of the object it is moving through (*i.e.* elevator, duct, pipe, etc.). Typically, holdup is difficult to quantify completely and accurately because there are literally tens of kilometers of ventilation and

vacuum lines in a plant such as the UMP. The goal of the team from Los Alamos National Laboratory (LANL) was to measure holdup, train Ulba personnel to make holdup measurements using the Generalized Geometry Holdup (GGH) assay technique, and assess additional requirements for an effective long-term program of holdup measurements at the plant.

1.2. Methodology of holdup measurements taken at Ulba

1.2.1. Overview of the GGH method and models

The GGH method is described in detail elsewhere [2]. The following is a brief overview of the technique and its models. Holdup measurements are usually made with collimated, shielded gamma-ray detectors. The distance between the deposit and detector is chosen to approximate the geometry of the deposit to that of an ideal point, line, or area in a radially symmetric field of view. The quantitative result is based on the net count rate of a representative gamma ray from the isotope of interest. Several corrections are applied to the measured rates, including those for the effects of room background and equipment attenuation effects. The GGH technique employs relatively new generalized procedures that correct for the effects of both finite-source dimensions and gamma-ray self-attenuation. The finite-source effect arises from the fact that point and line deposits are not ideal in that actual deposits have finite (non-zero) width. Ignoring the finite width results in a negative bias in the isotope mass determined by a calibration that assumes an ideal deposit geometry. The self-attenuation correction is required because uranium is highly attenuating to its own gamma radiation. The self-attenuation effect increases non-linearly with deposit thickness and also results in a negative bias if ignored. Because of the non-linear dependence on the pre-correction deposit thickness, the self-attenuation correction must be applied last. One parameter provided by the user, the “Finite-source Dimension”, applies to both corrections. This is the user’s best guess of the width of a point or line deposit. After all corrections are made to the data a specific isotope mass is obtained in units of grams for a point, g/cm for a line, or g/cm^2 for an area deposit. The total isotope mass of a line or area deposit is the specific mass multiplied by the deposit length or area, respectively.

1.2.2. Measuring LEU holdup

The 186-keV gamma ray is utilized for uranium holdup measurements because it is a direct measure of ^{235}U and because it has a moderate yield (~43,000 γ/s/g ^{235}U). It is often necessary to use the 1001-keV gamma ray to quantify uranium in LEU holdup. Three major problems arise with this gamma ray:

1) Its yield is very low (~75 γ/s/g^{238}U).
2) The photoelectric absorption probability in a 5-cm-thick sodium iodide (NaI) crystal is only 1.3% compared to 91% for the 186-keV gamma ray. This also results in a significant Compton continuum at lower energies.
3) The 1001-keV-gamma ray is highly penetrating, rendering the detector shield much less effective and resulting in the need to measure and subtract a substantial room background at each measurement location.

However, much thicker deposits can be measured with the 1001-keV gamma ray than with the 186-keV gamma ray. A deposit must be less than infinitely thick to the gamma ray in order to quantify the deposit in a strictly a passive measurement . Table I shows limiting thicknesses for the 186 and 1001-keV gamma rays for different uranium materials quantified using the passive gamma ray measurement.

material	density: ρ g/cm^3	186-keV gammas μ (186) cm^2/g	1/(μρ) cm	1001-keV gamma μ (1001) cm^2/g	1/(μρ) cm
metal	18.7	1.47	0.04	0.075	0.71
UF_6 (solid)	4.7	1.03	0.21	0.070	3.04
UO_2 (sintered)	10.9	1.31	0.07	0.074	1.24
UO_2 (powder)	2.0	1.31	0.38	0.074	6.76
U_3O_8 (highly packed)	7.3	1.26	0.11	0.073	1.88
$UO_2(NO_3)_2 \cdot H_2O$	2.8	0.76	0.47	0.071	5.03

Table I. Limiting thicknesses for various uranium materials.

These thicknesses were derived by taking into account the self-attenuation correction for a purely passive measurement. This correction takes the form of equation (1):

$$CF_{ATTEN} = const * \ln[1-\mu\rho x] \qquad (1)$$

where x is the thickness of the deposit. Thus, x cannot be greater than 1/(μρ) in order to evaluate the natural logarithm in the correction. Deposits at the UMP ranged from thicknesses thin enough to use 186-keV gammas to those for which even 1001-keV gammas were severely self-attenuated. While the 186-keV gamma ray has the advantage of better counting statistics, a combination of large attenuation effects and the presence of discrete interferences at the lower gamma-ray energy, as discussed below, enforces use of 1001-keV for quantitative analysis in many if not most situations in which LEU is measured.

1.3. Equipment used

The following equipment was used in each of three setups for measurements of holdup performed at the UMP in July 2004:

- shielded, compact NaI detectors (2.5-cm diameter by 5-cm-thick crystals)
- Canberra InSpector 2000 MCAs w/ Genie 2000 software
- Laptop computers with GGH specific spreadsheets
- poles and hardware to extend detectors to equipment for measurements
- tungsten plugs

The equipment for three setups used for measurements of holdup performed in October 2004 was basically the same with the exception of the MCAs (GBS Elektronik MCA166) and acquisition software (WinSPEC). Figure 1 is a spectrum of a relatively thick deposit acquired during the campaign in October showing the regions of interest (ROIs) as red dashed lines. The tungsten plug is inserted into the detector collimator to measure the room background contribution at each measurement location. Therefore, two spectra were acquired at each measurement location, one without and one with the tungsten plug. The relatively large room background contribution at 1001-keV in Figure 1 emphasizes the difficulty of measurements performed at 1001-keV and the importance of performing the measurements of room background.

The 186 and 1001-keV gamma ray peaks are clearly visible in Figure 1. Also of great interest is the small peak at 238-keV from the decay of ^{212}Pb in the ^{232}U decay chain. The presence of ^{232}U indicates that this is recycled uranium. Because the 238-keV peak in NaI spectra is barely resolved from the 186-keV peak, ROIs for the peak and continuum must be set conservatively, as illustrated, to assure that 238-keV activity does not contribute to the continuum ROI above the 186-keV peak. The ROI boundaries for the 186-keV peak and continuum were: 159 – 199-keV and 201 – 210-keV respectively. The 1001-keV peak and continuum ROIs were: 943 – 1069-keV and 1075 – 1201-keV respectively. Liberal ROI limits are possible at 1001-keV because of the absence of interferences. Such settings are much more forgiving of gain drift.

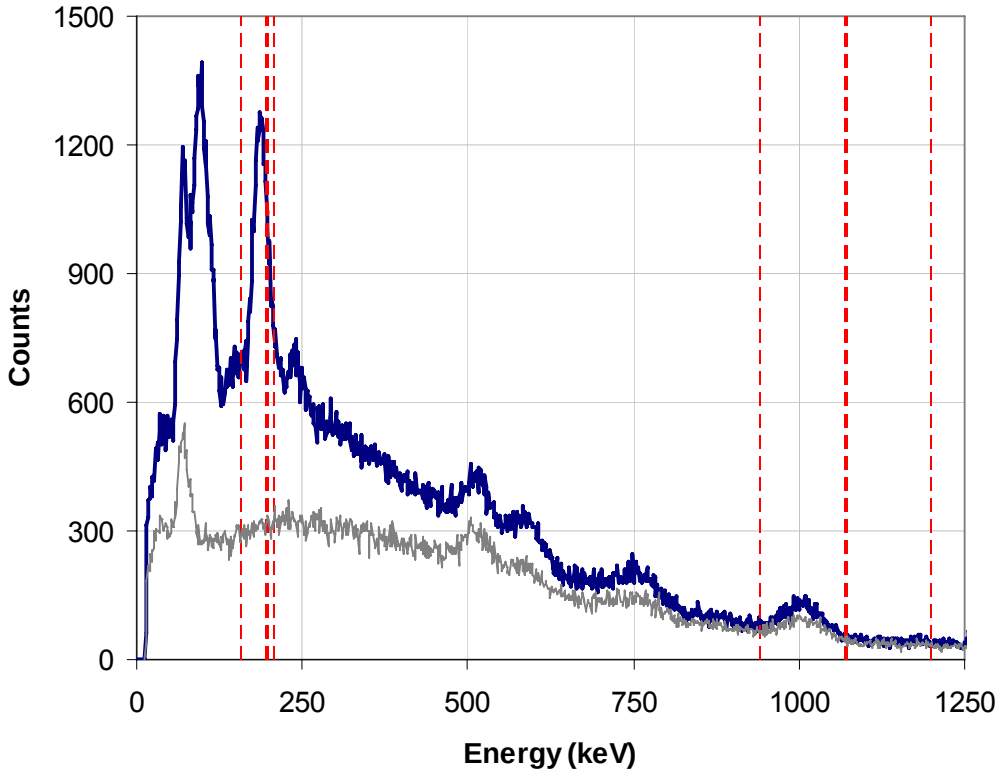

Figure 1. Spectrum taken of a ventilation line. The red dashed lines show the boundaries of the regions of interest (ROIs). The grey spectrum is room background measured for the same count time with the tungsten plug inserted in the collimator.

2. Data management

2.1. Calibration

Ulba processes LEU, and LEU holdup deposits are often (while HEU holdup deposits are seldom) infinitely thick to the 186-keV gamma ray, therefore, data for 186- and 1001-keV gamma rays must be acquired simultaneously. Likewise, the detectors must be calibrated at both energies. Two types of measurements at each energy were performed at LANL before each trip to calibrate the holdup measurements. The first uses a point source reference standard at a well defined distance and at the center of the detector's field of view (on-axis). This is an absolute calibration that assumes a $1/r^2$ dependence of the detector response to a point source. The second type of measurement establishes the relative response of the detector to off-axis locations of the deposit within the field of view. This is necessary to apply the absolute calibration to extended (line and area) deposit geometries. The relative response is constructed by measuring a point source at discrete locations off the axis of the detector, and since it is a relative (-to-the-on-axis) measurement, it is not necessary that the source be a standard. It is necessary that the energy of the gamma ray be the same, or nearly so, as that measured for the absolute calibration. The relative off-axis responses were measured using the 186-keV gamma ray from a ^{235}U source and the 1064-keV gamma ray from a ^{207}Bi source.

2.2. Measurement raw data

Raw data for each measurement were handwritten on a form that had spaces for the location identifier, a description of the measurement location, container material and thickness, count time, ROI integrals for the 186- and 1001-keV peak and continuum without the tungsten plug, ROI integrals for the 186- and 1001-keV peak and continuum with the tungsten plug, the source-to-detector

distance, a guess at the finite-source dimension, the deposit geometry (point, line, or area), the length of the line deposit or area of the area deposit, and any comments about the measurement. The count times for the measurements with and without the tungsten plug were set to be the same.

2.3. Holdup data analysis

A spreadsheet was utilized with an embedded Visual Basic (VB) program to analyze the measurement data. There are two worksheets in this spreadsheet, one for 186-keV (^{235}U) data, and the other for 1001-keV (^{238}U) data. In addition to the information from the raw data sheets described above, the spreadsheet requires the weight fraction of the isotope. Once all required information is entered in the spreadsheet a button is clicked that executes the VB program. This program takes the room-background-subtracted net count rate for the gamma-ray of interest, corrects for container wall attenuation, and then calculates a specific mass for that isotope using the appropriate calibration constant and distance multiplier, if any. This specific mass, which has units of grams for a point deposit, g/cm for a line deposit, or g/cm^2 for an area deposit, is then first corrected for finite-source effects and second for self-attenuation[2]. The program then calculates the total uranium holdup mass by dividing the corrected specific mass by the weight fraction of the isotope and multiplying by length of the extended line or by the area of the extended surface as appropriate. The program propagates the random uncertainty from counting statistics through the analysis algorithms to give the random uncertainty (σ) in the holdup mass. The program also performs tests on the data to determine if the deposit is thicker than or within 3σ of the limiting thickness. If the deposit is greater than the limiting thickness then the self-attenuation correction cannot be performed, and no final mass is calculated for that measurement. If the deposit is thin, but is within 3σ of the limiting thickness, the user should be cautious of the final results because the self-attenuation correction for these deposits is likely to be large and steeply changing with thickness, causing a sharp increase in the relative measurement uncertainty propagated through the correction algorithm.

Typically for extended line and area deposits, several measurements are taken at different locations. A specific mass is calculated for each measurement that is entered in the spreadsheet. The spreadsheet can average the specific masses for the extended line or area deposit and assign that average to the entire deposit. By averaging several measurements of the same extended deposit the final relative uncertainty in the average is smaller than those of the individual measurements. Short counts allow measurements at more deposit locations, providing better sampling of the extended deposits. The statistics for one long count are equivalent to those for the average of many short counts that sum to the same count time.

2.4. 186-keV vs. 1001-keV data

Analysis of the data revealed many locations where deposits were greater than the limiting thickness or nearly so to the 186-keV gamma ray. Because most of the measurements made at Ulba were of ventilation or vacuum lines this discussion refers to line deposits. The specific masses of deposits were typically non-uniform as a function of position. Furthermore, some locations in a given segment of process equipment might be greater than the limiting thickness to 186-keV gamma rays while others in the same segment might not. It is preferable, when the gain is stable, to use the 186-keV data to determine the mass of thin deposits because the counting statistics will be better. However, when the deposit is thick, the user must rely on the 1001-keV data and tolerate a larger uncertainty from poorer counting statistics. Nevertheless, statistics at 1001-keV improve for thicker deposits.

2.5. Verification and measurement control

At Ulba, each detector's absolute calibration was verified daily using HEU and depleted uranium (DU) sources of known uranium mass and isotopic composition provided by the facility. The radial response of the detectors was not verified at Ulba because of time considerations. The calibration verifications were all within 14% except for one detector whose calibration at 186-keV was low by 45%. This was later shown to be the result of a gain drift.

Each morning the 186-keV peak count rate, centroid (gain), and full width at half maximum (FWHM) were measured using the Ulba HEU source to ensure a consistent response from day to day. Each team also routinely verified the gain and FWHM after every 12th measurement in the plant. Some drift was observed, and appropriate gain corrections were made.

3. Vacuum line measurement results

Due to the sensitivity to publishing actual measurement results, the holdup masses in this report have been normalized.

3.1. VL39-49

The contents of vacuum line 39-49 are static. This line has been measured several times with the GGH method utilizing low-resolution detectors (NaI) and the 1001-keV gamma ray. It has also been measured multiple times[3] using ISOCS[4], a method that employs a high-resolution Ge detector. The line itself is located along the furnace hall and is 74 m long with a diameter of 9 cm. The measurement distance of 40 cm was suitable for GGH analysis of line deposits in this case. The distance between measurement points was chosen to be 1 m. The line was measured by three groups in July 2004 and by two groups in October 2004 using the GGH method. It was also measured by the IAEA twice (in September 2003 and in September 2004) with ISOCS. Ulba personnel also measured the line with their ISOCS system in September 2004. The total uranium mass result and 1σ random uncertainty for each measurement are given in Table II. It should be noted that the approach used with the ISOCS system by the IAEA and Ulba was one of verification of the presence of material. They have purposefully implemented a conservative quantification algorithm and anticipate a better understanding of the facility holdup as they gain more experience.

Group	Date	Normalized Uranium Mass	Random 1σ Uncertainty in Normalized Uranium Mass
GGH-A	July 2004	1.00	0.21
GGH-B	July 2004	0.87	0.17
GGH-C	July 2004	0.73	0.43
GGH-A	October 2004	0.75	0.05
GGH-B	October 2004	0.96	0.06
ISOCS-Ulba	September 2004	0.48	none reported
ISOCS-IAEA	September 2003	0.57	none reported
ISOCS-IAEA	September 2004	0.38	none reported

Table II. Normalized uranium mass results for VL39-49.

Because the deposits were greater than the limiting thickness to the 186-keV gamma ray at many measurement points along this vacuum line, masses reported using the GGH method are determined from 1001-keV data. The average correction for self-attenuation was 10 – 15% for the five sets of data, and other corrections are considerably less. All the masses reported from the GGH measurements agree with one another at the 2σ level for random uncertainty, and four of the five agree at the 1σ level with the average of all five measurements being 0.86 ± 0.10 where the uncertainty is the propagated random uncertainty in the mean. The 1σ standard deviation of the five GGH results is 0.12. The average of the ISOCS results is 0.48 with a standard deviation of 0.09 for the three measurements. The ISOCS measurements were typically a few points along the vacuum line, but for many hundreds of seconds each, thus, the random uncertainties are negligible compared to the systematic uncertainties.

Plotting the normalized specific mass as a function of measured distance or linear position along the process equipment can be useful. Such maps reveal where holdup accumulates faster relative to other points along a line. The user can also compare equivalent maps over time to and possibly predict when and where clean-outs will be required. Figure 2 shows the specific mass as a function of distance for VL39-49 as measured by group GGH-A in July 2004. Note larger error bars at locations of thicker deposits caused by blow-up of the relative uncertainty propagated through the self-attenuation correction algorithm.

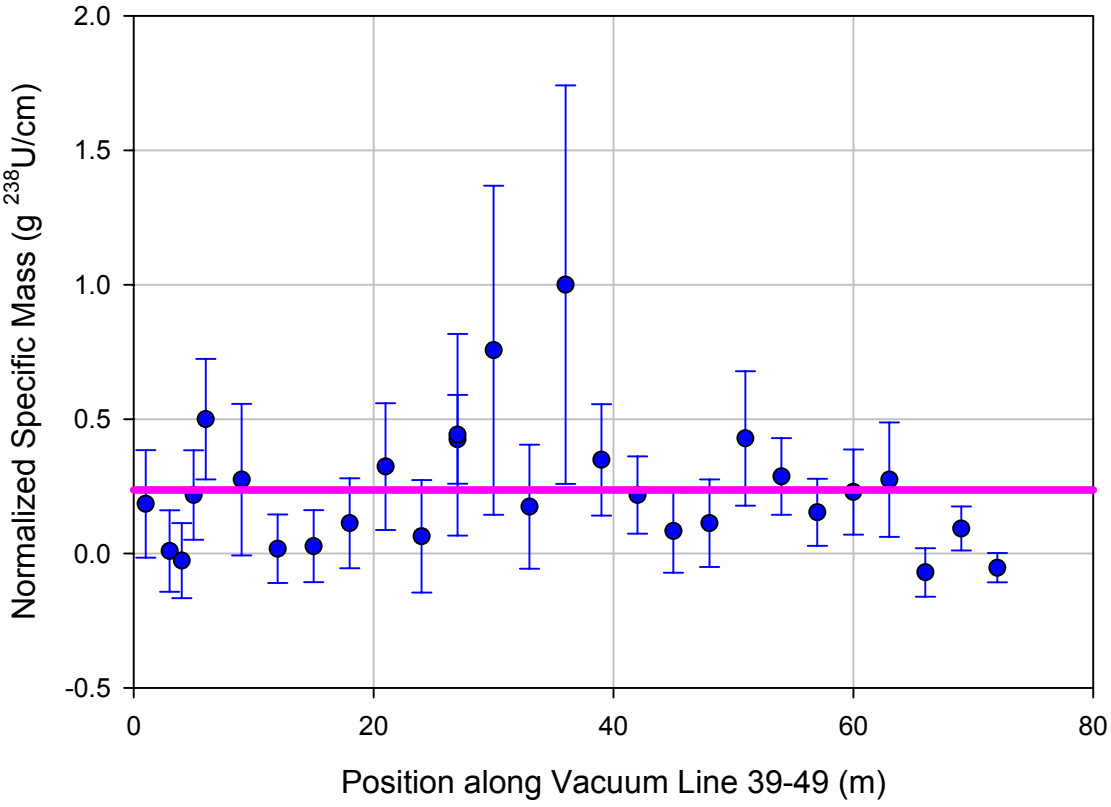

Figure 2. Normalized specific mass of ^{238}U vs. position along VL39-49. The pink line indicates the average normalized specific mass (0.236 g/cm).

3.2. VL36-42

This vacuum line was located along a corridor and above the ceiling panels, like VL39-49, the contents were static. Additionally, there is a second process pipe below VL36-42 that may have had material in it. This intervening pipe increases the wall attenuation correction factor from 1.29 to 2.16. The line is 36 m long and 9 cm in diameter. This line was measured once with the GGH method in July 2004 and with the ISOCS method in September 2003 [3]. The GGH data showed that the deposit was greater than the limiting thickness to the 186-keV gamma ray at all measurement locations, and also greater than the limiting thickness to the 1001-keV gamma ray at several locations. Furthermore, deposits at many locations were within 3σ of the limiting thickness to the 1001-keV gamma ray so that the measurement uncertainty is very large in these cases. Self-attenuation corrections were performed for all locations for which the deposits were not greater than the limiting thickness, and the corrected results used to determine the average for the line. The average self-attenuation correction is 1.417 with a maximum correction of 2.074. Deposits too thick to be corrected for self-attenuation were not included in the average. This circumstance causes the GGH results for uranium mass in VL36-42 given in Table III to be suspect (biased low). The material profile is shown in Figure 3.

Group	Date	Normalized Uranium Mass	1σ Normalized Mass Uncertainty
GGH	July 2004	1.0	0.44
ISOCS	September 2003	0.60	none reported

Table III. Total uranium mass results for VL36-42.

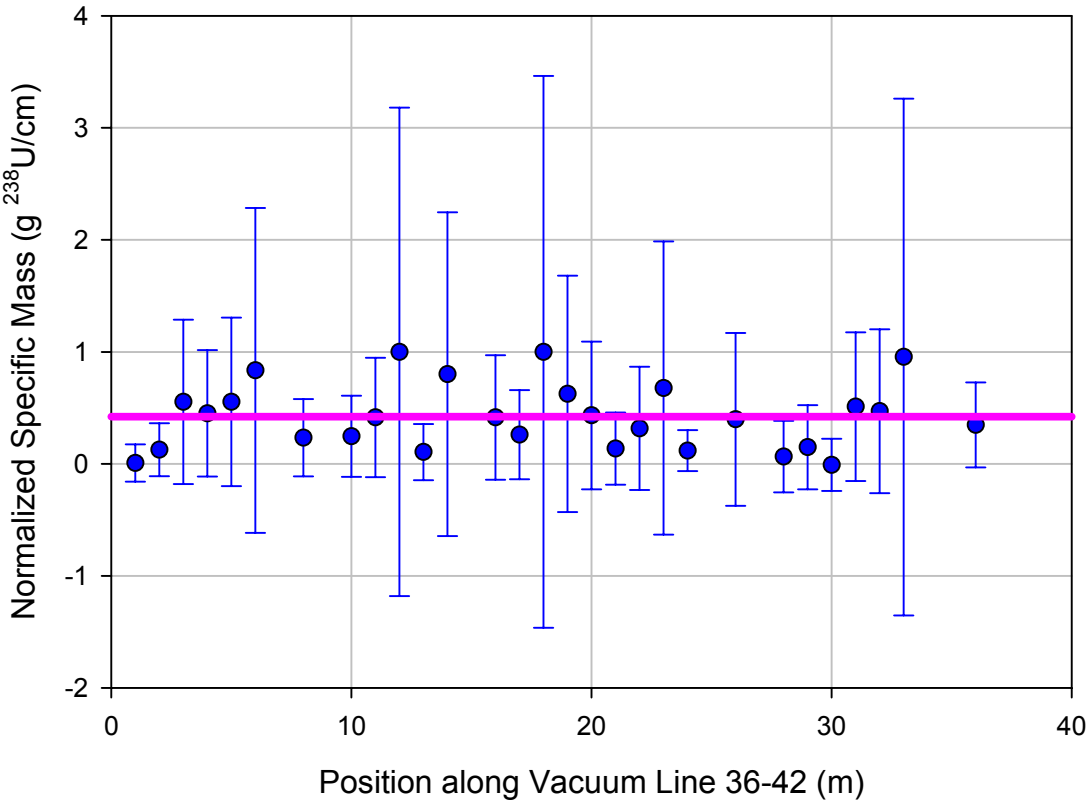

Figure 3. Normalized specific mass of ^{238}U vs. position in Vacuum line 36-42. The pink line indicates the average specific mass (0.42 g/cm).

4. Ventilation line measurements

Much more detail is given in the final report [5] generated from these measurement campaigns. The report includes diagrams of the ventilation lines as well as material profiles, like those shown in Figures 2 and 3, for all the segments measured in building 600 of the UMP. The facility provided schematic diagrams of the ducting and vacuum lines that were used to mark on to identify the measurement positions. These diagrams are organized by ventilation system (VG8, VG6, or VG52) and are labeled with a unique "Sheet" number. An example is given in Figure 4; this is "Sheet 1-1-1". Each unique segment on a sheet was assigned a unique color for ease of displaying the results. The results of the measurements taken in July and October of 2004 will be summarized in a table for each ventilation system. The tables will reference "Sheet" numbers and segment colors that correspond to the facility drawings that detail specific areas of that ventilation system, as shown in the final report.

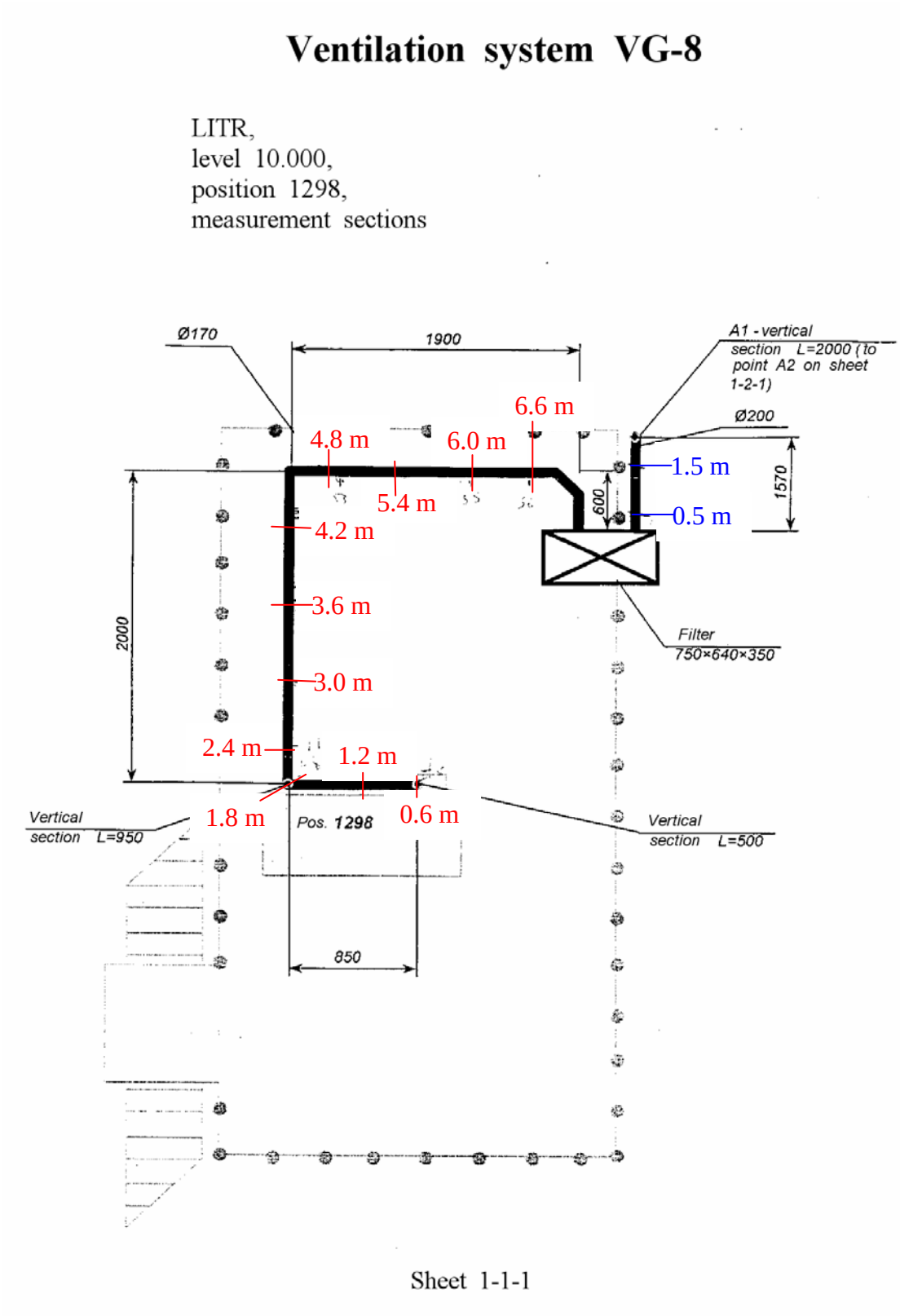

Figure 4. Example of the schematic drawings provided by the facility showing the arrangement of the ducting and the points where holdup was measured.

4.1. VG8

The plan for the July 2004 campaign was to measure the VG8 ventilation system because it is the most accessible and contains a significant quantity of material. Three groups, working independently and simultaneously, each measured different segments of VG8. The results for all segments were summed to give a final total mass for all of the VG8 system that was measured. Some segments of VG8 that were inaccessible or impractical to measure because of time constraints are not included in this total for VG8.

Table IV gives a summary of the uranium holdup measured in VG8 separated by Sheet number and segment. The table includes the normalized uranium mass in the segment and its random uncertainty, the length of the duct or area of equipment, the gamma-ray data used in the calculations, and the average finite-source and self-attenuation corrections.

Sheet(s)	Segment	Normalized U Holdup	Normalized Uncertainty	Length of Segment (m) or Area of Equip. (m^2)	Gamma-ray Measured (keV)	Average Finite Source Correction	Average Self-Attenuation Correction
4-1-1	red	6.2E-02	1.5E-02	8.0	1001	1.240	1.055
4-1-1	green	2.6E-02	8.4E-03	1.8	1001	1.450	1.140
4-1-1	blue	5.7E-02	1.3E-02	4.0	1001	1.200	1.112
1-1-1	red	5.1E-05	1.2E-04	5.4	186	1.170	1.008
1-2-1 & 1-1-1	blue	6.3E-04	3.0E-04	9.9	186	1.218	1.022
1-2-1	red	1.0E-04	1.7E-04	1.2	186	1.098	1.049
1-3-2 & 1-2-1	green	3.9E-03	1.2E-03	3.6	186	1.083	1.411
1-3-2	blue	3.5E-03	3.0E-03	2.2	186	1.197	1.681
1-3-2 & 1-3-1	red	8.4E-04	5.1E-04	14.3	186	1.108	1.028
1-3-1	purple	6.8E-04	3.4E-04	5.4	186	1.047	1.069
1-3-1	black	3.4E-04	1.7E-04	3.8	186	1.047	1.063
1-4-1	red	9.6E-03	1.0E-03	17.0	186	1.111	1.190
2-1-1 Level 4	red	8.3E-03	8.4E-04	2.2	186	1.136	1.135
2-1-1 Level 4	blue	1.5E-02	2.7E-03	7.7	186	1.114	1.064
1-5-1	blue	1.3E-03	5.6E-04	4.2	186	1.098	1.138
1-5-1	red	1.1E-03	6.3E-04	2.7	186	1.106	1.165
1-5-1	green	6.8E-04	8.4E-04	3.6	186	1.098	1.069
1-5-2	red	2.3E-02	7.9E-03	15.6	186	1.067	1.361
1-5-2	blue	3.9E-03	2.5E-03	1.1	186	1.076	1.669
1-5-2	green	1.2E-03	1.2E-03	1.1	186	1.045	1.202
1-5-2	black	2.9E-03	1.0E-03	2.3	186	1.000	1.060
1-5-2	purple	2.6E-02	3.4E-02	1.6	1001	1.043	1.300
2-1-1 Level 2	red	2.5E-03	5.1E-04	14.0	186	1.250	1.034
1-6-1	red	2.2E-03	6.8E-04	17.0	186	1.056	1.043
1-6-1	blue	4.6E-03	1.4E-03	14.5	186	1.054	1.052
1-6-1 & 1-5-1	green	6.8E-04	5.1E-04	7.0	186	1.026	1.243
3-1-1	red	1.0	0.12	32	1001	1.086	1.285

Table IV. Summary table for VG8 measurements. The segments highlighted in blue were measured twice by different groups.

4.2. VG6

Segments of VG6 were measured during both the July and October 2004 trips. VG6 was not measured to the same extent as VG8. Table V gives a summary of the uranium holdup measured in VG6 separated by Sheet number and segment. The table includes the normalized uranium mass in the segment and its random uncertainty, the length of the duct or area of equipment, the gamma-ray data used in the calculations, and the average finite-source and self-attenuation corrections.

Sheet(s)	Segment	Normalized U Holdup	Normalized Uncertainty	Length of Segment (m) or Area of Equip. (m^2)	Gamma-ray Measured (keV)	Average Finite Source Correction	Average Self-Attenuation Correction
9-4-1		0.03	0.01	1.0	186	1.028	1.145
9-4-1		0.00	0.01	0.8	186	1.028	1.028
9-5-1	red	0.08	0.02	3.9	186	1.228	1.041
9-5-1	blue	0.12	0.04	3.9	186	1.017	1.155
9-5-1	green 67-69	0.09	0.05	7.0	186	1.032	1.062
9-5-1	green 65	0.00	0.00	0.2	186	1.017	1.000
9-5-1	green 63-64	0.07	0.02	1.6	186	1.088	1.108
9-6-1	blue 1-6	0.17	0.04	2.1	186	1.048	1.213
9-6-1	blue 9-11	0.01	0.00	0.5	186	1.120	1.057
9-6-1	blue 17-19	0.00	0.01	2.3	186	1.046	1.009
9-6-1	red	0.49	0.22	2.5	186	1.053	1.621
9-6-1	green	0.69	0.12	12.0	186	1.053	1.163
9-6-1	black	0.03	0.02	1.4	186	1.028	1.106
9-6-1	purple	0.01	0.02	1.8	186	1.021	1.045
9-6-2	blue	1.00	0.28	6.2	186	1.078	1.512
9-6-2	red	0.09	0.03	1.8	186	1.021	1.331
9-6-2	green	0.07	0.03	4.9	186	1.065	1.038

Table V. Summary table of VG6 measurements. The segments highlighted in blue were measured twice by different groups.

4.3. VG52

Similar to VG6, segments of VG52 were measured during both the July and October 2004 trips. VG52 was not measured to the same extent as VG8. Table VI gives a summary of the uranium holdup measured in VG52 separated by Sheet number and segment. The table includes the total uranium mass in the segment and its random uncertainty, the length of the duct or area of equipment, the gamma-ray data used in the calculations, and the average finite-source and self-attenuation corrections.

Sheet(s)	Segment	Normalized U Holdup	Normalized Uncertainty	Length of Segment (m) or Area of Equip. (m^2)	Gamma-ray Measured (keV)	Average Finite Source Correction	Average Self-Attenuation Correction
8-1-1	red 1 & 8	1.6E-03	6.2E-03	2.3	186	1.068	1.066
8-1-1	red 2 - 7	5.5E-04	5.4E-03	7.3	186	1.094	1.041
8-1-1	red 10 & 11	5.5E-03	5.2E-03	2.6	186	1.094	1.141
8-1-1	green	-2.4E-02	4.5E-02	2.6	1001	1.137	1.000
8-1-1	blue	4.2E-02	2.5E-02	2.5	1001	1.137	1.050
8-1-2	red	0.1	8.5E-02	4.9	1001	1.017	1.186
8-1-2	blue	4.6E-02	4.4E-02	5.2	1001	1.034	1.078
8-1-2	black	3.3E-03	8.8E-03	0.7	1001	1.034	1.028
8-2-1	blue	2.4E-02	7.7E-03	10.3	186	1.094	1.139
8-4-1	red	2.5E-03	2.6E-03	1.9	186	1.094	1.159
8-5-1	green	2.1E-02	7.7E-03	6.2	186	1.094	1.199
8-5-1	blue	1.0	3.3E-01	10.7	1001	1.190	1.256
8-5-1	red	0.9	2.6E-01	10.7	1001	1.137	1.277

Table VI. Summary table of VG52 measurements. The segments highlighted in blue were measured twice by different groups.

5. Analysis of Systematic Error

Certain parameters must be estimated by the user in order to arrive at a final value for the mass. The uncertainty in these parameters affects the ultimate confidence in the final holdup mass. Therefore it is important to be able to quantify these systematic effects.

Because the algorithm used to correct specific mass for self-attenuation is non-linear with respect to the uncorrected specific mass, the correction and the associated propagation of uncertainty must be performed measurement by measurement after other corrections have been made. Statistics dominate the uncertainty of an individual holdup measurement performed in a short count time, and these are accordingly propagated, independently for each measurement. While the systematic effects also propagate uniquely for each measurement, the magnitude of the effect on each measured quantity tends to be the same (unlike the magnitude of the statistical effect). Therefore, it is appropriate to recalculate the overall holdup result with each systematic parameter variation (e.g. the ^{235}U fraction may be 3.0 ± 0.5% for all measurements) to obtain the systematic contribution to the uncertainty from each parameter. We, therefore, advocate obtaining a numerical estimation of the systematic effects on the final mass by variation of each user-specified parameter and repeated execution of the analysis tool.

6. Conclusions and Future Work

The collaborative effort between UMP, IAEA, and LANL completed measurements of approximately one third of the ventilation and vacuum systems and approximately four fifths of the holdup inventory inside building 600 at UMP. Using low-resolution, portable, room temperature gamma-ray spectroscopy systems and GGH analysis tools, the uranium holdup was measured in UMP ventilation and vacuum lines with a 1σ random uncertainty of 12%. Repeated measurements of VL39-49, and one segment each of VG8, VG6, and VG52, in July and October 2004 using different measurement equipment but the same method (GGH or ISOCS) show good agreement. The use of the two measurement techniques (GGH and ISOCS) at Ulba are intended to be complimentary. The more portable method (GGH) is intended to access most of the ventilation and vacuum systems that the ISOCS system cannot. Conversely, the ISOCS technique is better suited to more complex geometries such as process components and waste.

The measurements performed in two campaigns to the UMP are the first to achieve or demonstrate the following.

- Low-resolution quantitative gamma spectroscopy of holdup in a wide dynamic range (100 –1000-keV).
- Highly portable quantitative measurements of very thick deposits of uranium oxide.
- Practicality of measuring 1001-keV gammas from a large holdup deposit in a “sea” of highly penetrating 1001-keV gammas from room background.
- Viability of the portable quantitative measurements for LEU holdup on the scale of the Ulba Plant.
- Implementation of the generalized holdup correction algorithms for LEU.
- Broad understanding of the range of self-attenuation effects in an LEU-solids plant, and the assurance that the gamma techniques are suited to this range.
- Assurance of the accessibility of most UMP deposits to measurements performed using highly portable gamma spectrometer detectors.

The measurements reinforce the need for the following procedures and capabilities – many unique to LEU measurements – for successful routine implementation of portable low-resolution gamma-ray measurements at UMP:

- A spectrometer detector with good detection efficiency through 1001-keV.
- Simultaneous measurements of both 186- and 1001-keV gammas.

- A fixed internal reference source of ^{241}Am for measurement control spectrum-by-spectrum and gain-drift compensation.
- Alternative to ^{238}U source for timely measurements of off-axis detector response during calibration.
- A measurement, with the tungsten plug inserted, of room background for every deposit measurement.
- Conservative setting of the ROI on the 186-keV peak to avoid the effects of interference from the 238-keV gamma.
- Full automation of the data acquisition and measurement control. This will result in a measurement time savings of an order of magnitude [6].
- Stabilization against or compensation for scintillator gain drift.
- Alternative geometric models for self-attenuation.
- Implementation of (an energy-dependent) mean gamma interaction depth in the crystal.
- Full access by users to all input parameters used by the analysis tool to assure ability for numerical determination of systematic effects.
- Implementation of numerical evaluation of systematic contributions to measurement uncertainty.
- Flexible implementation of analysis for either (186- or 1001-keV) depending on equipment cleanout status.
- Implementation of the 3σ warning in approach to thick deposits.
- RFID tagging and readout of holdup measurement locations.

The UMP/IAEA/LANL collaboration is committed to implementing a viable portable quantitative capability for routine measurements of LEU holdup at Ulba.

7. Acknowledgements

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8. References

1. http://www.ulba.kz/

2. Russo, P. A.; *Gamma-ray Measurements of Holdup Plant-wide: Application Guide for Portable Generalized Approach*; Los Alamos National Laboratory report LA-14206; March 2005.

3. Lebrun, A.; Private correspondence; March 2004

4. *In Situ Gamma Spectroscopy with ISOCS, an In Situ Object Counting System*; Canberra Application Note; http://www.canberra.com/pdf/Literature/m2352.pdf

5. Belian, A. P.; et. al., *Holdup Measurements at the Ulba Metallurgical Plant*; Los Alamos National Laboratory report (LA-CP# TBD contact A. Belian: abelian@lanl.gov); March 2005.

6. Russo, P. A.; Sprinkle, J. K.; Bjork, C. W.; McKown, T. O.; Sheppard, G. A.; Smith, S. E.; and Harris, J. F.; *Evaluation of the Integrated Holdup Measurement System with the M3CA for Assay of Uranium and Plutonium Holdup*; Los Alamos National Laboratory report LA-13387-MS; August 1999.

Improvements in the Accuracy of the U235 Measurement in Drigg PCM Waste Drums

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Abstract:

Accurate determination of the U235 content of PCM wastes generated during retrieval operations at Drigg is complicated by the potential presence of self-shielding lumps of uranium. The routine assay functions of the Drigg Drum Monitors are unable to determine whether uranium lumps are present, and hence pessimistic assumptions regarding the degree of self-shielding in any uranium present have been applied in the calibration, to meet criticality safety requirements. A significant number of drums have been measured as having U235 content greater than the limits defined in the criticality clearance certificates, however it is likely that the U235 content of these drums has been grossly overestimated because of the assumed self-shielding (which equates to approximately a factor of eight compared to infinitely dilute uranium). These "high U235 content" drums present significant operational problems, as the only solution is to manually repack the waste into several other drums to lower the fissile content per drum, thus allowing transport of the waste to the Engineered Drum Store at Sellafield.

To address this issue, BIL Solutions Ltd. has developed a technique for identifying whether lumps of uranium are present. If the uranium present can be proven to be dilute, then the U235 result is reduced by removal of the assumed level of self-shielding from the calibration. A novel application of the Differential Peak Absorption technique is used to examine the relative intensity of different energy gamma rays from U235, after correction for matrix attenuation using a transmission source measurement, and determine whether the uranium is present as dilute or lumpy material. This technique can be used alone to determine whether lumps are present, or can be supported by a further novel technique utilising the self induced fluorescence X-Rays from uranium. In this application both techniques have been deployed.

To support the introduction of this technique and to determine whether lumps of uranium are genuinely present in the waste, twelve of the highest U235 content drums were selected for physical repacking. These drums were opened and individual waste packets removed and physically inspected to look for evidence of lumpy material. The packets were then placed within new drums and re-assayed to determine their fissile content. Additional measurements were then performed of the highest U235 content packets from each original drum, to obtain high quality gamma spectra for the analysis. The analysis results indicated that two of the twelve drums contained significant lumps of uranium, with the remaining ten being dilute.

The waste measurement results presented in this paper demonstrate the applicability of this technique, which will now be applied to future high U235 content PCM waste drums to improve the accuracy of the U235 measurement and to avoid unnecessary drum repacking.

Keywords:waste; U235; self-shielding; accuracy

1. Introduction

As part of the Plutonium Contaminated Material (PCM) retrieval operations at BNFL's Drigg site, two combined Passive Neutron Coincidence Counting (PNCC) and High Resolution Gamma Spectrometry (HRGS) drum monitors supplied by Canberra-Harwell Ltd. are used to assay PCM waste contained in 200 and 400 litre drums. A significant proportion of the waste drums are also contaminated with uranium, and the drum monitors are required to determine the total fissile content (i.e. total Pu + U235), to demonstrate compliance with criticality safety limits for drum transport and storage. The drums are then transported by rail to an Engineered Drum Store at Sellafield for safe surface storage. The U235 content of the waste drum is measured using the principal gamma emission at 186keV. The two drum monitors comprise of a neutron measurement chamber together with an HRGS system, which performs a segmented scan of the drum. This segmented scan, which includes a transmission source measurement at each of 16 vertical segments, facilitates more accurate quantification of the activity of any gamma emitting nuclides present in the drum.

Lumps of uranium exhibit gamma self-shielding. This results in a significantly lower rate of 186keV photons escaping per gram of uranium than would be expected for finely dispersed uranium powder. As the drum monitors cannot correct for these gamma self-shielding effects, their U235 calibrations assume a pessimistic level of self-shielding (a factor of 8.33 compared to dilute uranium) to avoid underestimation of the U235 content. Over the past few years of PCM retrieval operations, a large number of drums have had measured U235 contents greater than the limits defined in the criticality clearance certificates. This has resulted in significant operational problems, since the only acceptable course of action is to physically repack the contents of the original drum into several new drums to reduce the fissile content per drum, thus allowing safe transport of the waste. This drum repacking is very time-consuming and results in significant dose uptake to the plant operators involved.

The U235 assay results are known to be significantly overestimated due to combination of several pessimistic calibration assumptions, the largest of which is the self-shielding. In 2004, a proposal was made to remove the self-shielding assumption, which was justified on the basis of the following points:

- Removal of the self-shielding assumption will result in significant underestimation if uranium lumps are present. However, criticality modelling has shown that in all credible lump geometries, the underestimation of the U235 mass due to gamma self-shielding is compensated for by a reduction in reactivity due to thermal neutron self-shielding effects. This means that the circumstances required for criticality to occur in Drigg wastes due to the underestimation of the U235 mass of uranium lumps are incredible.

- Other assumptions in the U235 calibration (e.g. matrix attenuation and source position) are significantly pessimistic under most credible scenarios, and hence would reduce or cancel out the assay underestimation in cases where lumps of uranium are present.

- Based on historical records, very few drums were sent to Drigg PCM stores with high uranium content, and it is considered highly unlikely that significant lumps of uranium will be present due to the commercial value of the material at the time of disposal.

To build further confidence in the historical records, a sample of 12 high U235 content drums were identified for physical repacking to confirm (or otherwise) the absence of lumps of uranium in the Drigg wasteform. This trial also gave the opportunity to validate the new uranium lump detection analysis developed by BIL Solutions Ltd.

2. Waste Drum Repacking Operations

Table 1 below provides details of the sample of 12 waste drums that were selected for physical repacking, which were chosen based on the following criteria:

- The drums have high U235 assay results and are therefore most likely to yield useful data regarding the presence of lumps.

- The majority will still require repacking even if the self-shielding pessimism is subsequently removed (i.e. the U235 mass is greater than 8.33× the transport limit).

- Information on the spatial distribution of U235 in the drum (i.e. from the 16 vertical segment scan) has been used to identify drums with more localised U235 concentrations. Such drums are considered more likely to contain uranium lumps than those showing a more uniformly dispersed U235 signal.

- The origins of the drums have been considered to ensure that a representative sample from the plants that were most likely to have consigned uranium bearing waste to Drigg.

Drum Number	Total Fissile (g)	Total Pu (g)	U235 (g)	Gamma Radiation (µSv/hr)	Drum Volume (l)
Drum 1	1564.06	0.74	1563.33	<2	400
Drum 2	1851.10	47.50	1809.60	<2	200
Drum 3	2325.26	4.20	2321.06	2	400
Drum 4	1035.06	6.60	1028.46	<2	200
Drum 5	2154.83	1.34	2153.49	2	400
Drum 6	1059.18	14.30	1044.89	4	200
Drum 7	878.40	17.40	861.00	8	200
Drum 8	2247.00	47.00	2200.00	10	200
Drum 9	1035.28	85.50	949.78	10	400
Drum 10	691.40	73.20	618.20	45	400
Drum 11	5350.00	98.00	5252.00	50	200
Drum 12	9449.61	89.50	9360.11	40	400

Table 1: Details of drums selected for repacking.

The repacking operations involved opening each drum and removing individual waste packets contained therein. The waste packets visually inspected to look for evidence of uranium lumps, weighed, the physical dimensions measured, the dose rates measured (gamma and beta/gamma) and photographs were taken.

The waste packets were then bagged in PVC and placed in new 200 litre drums, with the packets being approximately centred within the new drum (using PVC packaging) to improve the accuracy of subsequent assay. Where possible, one packet was placed in each new drum, however, in several cases, this was impractical due to the total number of packets in the original drum, and some drums containing two or more packets were generated. In total 199 packets were removed from 12 drums, and 168 new drums were produced, which is an average of 14 new drums / original drum. The new drums containing individual waste packets were then re-assayed with a drum monitor, to obtain revised U235 mass results. In addition, longer duration “Maintenance Mode” measurements were performed of the packets with a high U235 content, to improve the precision in the gamma spectra for offline analysis to look for evidence of uranium lumps.

3. Data Analysis Methodology

The aim of the measurements of the repacked drums is to improve the accuracy of the U235 measurement. This is done using a combination of Differential Peak Absorption (DPA) and Self Induced X-Ray Fluorescence (SIXRF) analyses to provide conclusive evidence that the uranium present is dilute and does not exhibit significant self-shielding. If this is found to be the case, then the self-shielding factor assumed in the calibration can be removed. A detailed description of these two

analysis techniques and a discussion on the accuracy of the current U235 measurement is provided below. Note that the DPA technique is the primary method used to determine whether the uranium is dilute, with the SIXRF technique being used to provide additional confidence.

3.1. Accuracy of Current U235 Measurement

The purpose of the U235 measurement is to demonstrate compliance with criticality safety limits for drum transport and storage. However, because of limitations in the measurement technique, several assumptions must be made in the uranium calibration to ensure that unsafe results are not reported. These assumptions tend to be pessimistic under general measurement conditions, and are discussed further below:

(i) Self-shielding. The U235 mass calibration assumes that any uranium present is in the form of a lump, with an 186keV self-shielding factor of 8.33. This factor was derived from Monte Carlo MCNP [1] modelling of the uranium calibration sources, and means that only 12% of the 186keV gamma signal reaches the outside of the lump. Approximately factor of eight overestimation will therefore arise if the uranium is present in dilute form (e.g. surface contamination).

(ii) Segment overlap. To achieve a good sensitivity, the gamma system was designed with an open collimator, with a vertical field of view on axis approximately 270mm high. However, to enable a more refined transmission correction factor the passive emission data is acquired at 16 vertical positions with a separation of 67mm. The calibration assumes that if a point source is present in the drum, it is only "seen" by the HRGS detector when it is measuring the segment in which the source is located. Any cross-talk between segments is assumed to be zero, however since the source is "seen" in adjacent segments, this will lead to an overestimation of the U235 mass. The amount of overestimation will be dependent on the true source distribution; however, it is estimated to be about a factor of three under typical measurement conditions.

(iii) Uranium distribution. The U235 mass is calculated assuming that all the U235 is present effectively as a point source at the centre of the drum (i.e. the position of lowest sensitivity), when applying the transmission correction. The amount of pessimism arising from this assumption will depend on the true distribution of uranium within the drum and the density of any waste matrix present. The overestimation increases as the matrix density increases, however, PCM wastes are generally low in density (typically 0.1 to 0.5g/cc) and overestimations of the order of 100% would therefore be expected. For the repacked drums, the waste packets were centred radially within the drum, and the matrix densities were generally low, so the pessimism arising from this component is expected to be insignificant compared to the two other assumptions.

It should be noted that these pessimistic assumptions are multiplicative, so for a drum containing dilute uranium, the overestimation of the U235 mass could be greater than a factor of 50 (i.e. 8 x 3 x 2).

3.2. Differential Peak Absorption (DPA) Analysis

The Differential Peak Absorption (DPA) technique involves examination of the count rates for gamma-rays of different energies emitted by the same radionuclide. Differences in the ratios of these count rates at different energies compared to the theoretical values gives a measure of the degree of attenuation of the gamma signal, since the attenuation is a function of gamma-ray energy. It is therefore possible to determine the attenuation of the gamma signal without the requirement for a transmission measurement or other correction. In the case of sources exhibiting self-shielding such a technique offers clear advantages over a conventional transmission correction, because the self-

shielding effects can be significant even in small lumps which would be unlikely to be “seen” in the transmission measurement.

However, this technique is only possible if the nuclide emits gamma-rays of sufficient intensity at different energies, and the greater the difference in energy the more effective the technique becomes. Furthermore, once the thickness of the lump exceeds a certain value (related to the mean free path of the gamma ray, which is dependent on its energy), the signal reaching the outside of the lump at that energy will remain constant as the lump size increases. This thickness is referred to as the “infinite depth” and the consequence is that the DPA technique can only provide a correction if the “infinite depth” has not been reached. At lump sizes greater than the “infinite depth”, the technique will indicate that “infinite depth” has been reached, but it is not possible to determine the total size of the lump.

The principal gamma rays from U235 are listed in Table 2.

Gamma-Ray Energy, keV	**Intensity, gammas / Bq**
143.8	0.10962
163.3	0.050814
185.7	0.5724
205.3	0.050112

Table 2: Principal gamma rays from U235 [2].

Although close in energy, the DPA technique can be applied to the different energy gamma-rays from U235 by a novel extension to the standard DPA technique. In order to illustrate how the technique has been applied to the Drigg data, Figure 1 shows the ratios of the U235 gamma-rays in different diameter lumps of uranium (of density = 2.1 g/cm3). These results have been determined using an empirical formula from [3]. Note that for higher density uranium, the ratios show the same trends but reach the constant values quicker (i.e. for smaller diameter lumps).

For infinitely dilute (i.e. non-self-shielding) uranium, the ratio of the 143/186keV gamma peaks is 0.192. This is the value obtained by simply taking the ratios of the intensities in Table 2. Similarly, the ratio of the 163/186keV gamma peaks is 0.089, and the ratio of the 205/186keV gamma peaks is 0.088. At a certain uranium lump diameter, the ratios reach a constant value, which is referred to as the infinite thickness. Above this diameter, a change in lump diameter does not alter the ratio. As can be seen, the constant values are 0.102 for the 143/186keV gamma peaks, 0.065 for the 163/186keV gamma peaks and 0.111 for the 205/186keV gamma peaks.

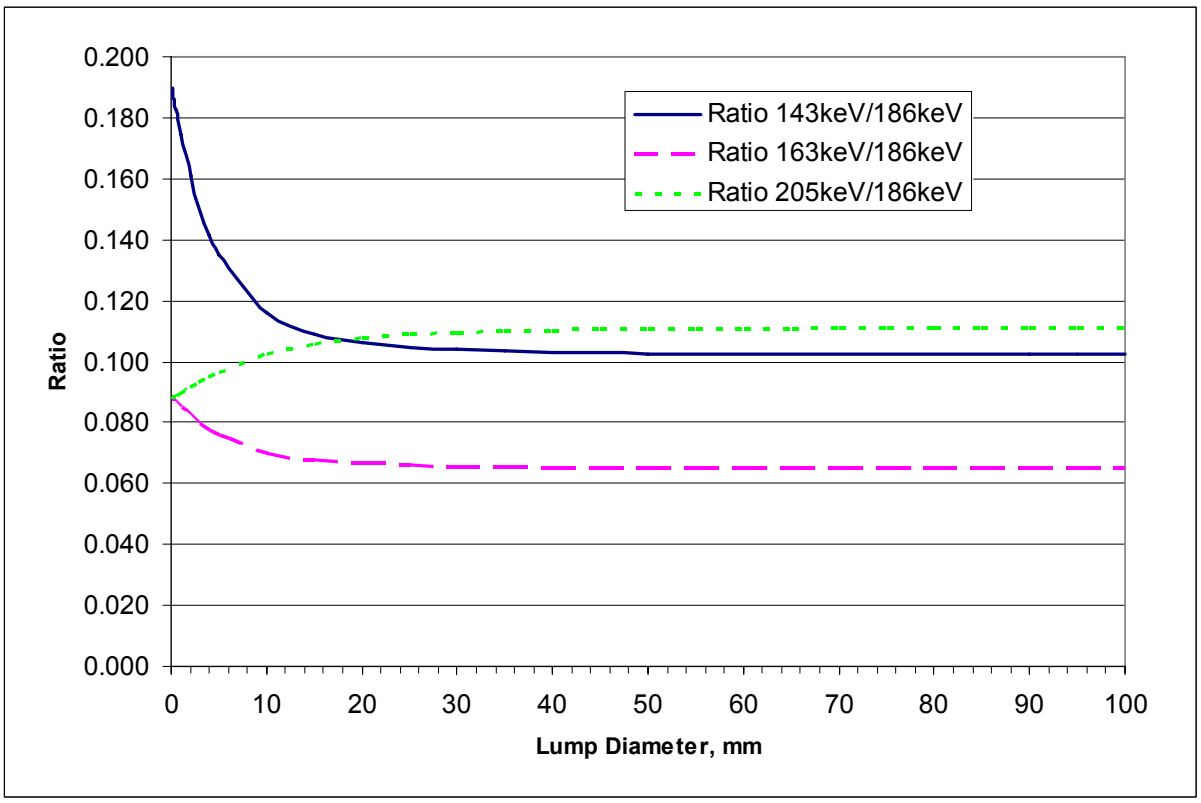

Figure 1: Variation of U235 peak ratios with lump diameter for spherical lumps (density 2.1 g/cm^3).

The peak ratios will also be affected by attenuation within the waste matrix. However, this will be a significantly smaller effect than the attenuation within the uranium lump, particularly for drums containing individual waste packets, since the waste matrix materials are generally much lower in density and atomic number. The B746 Drum Monitor performs a transmission measurement using a Eu152 source, and therefore it is possible to correct each of the U235 peaks for matrix attenuation. This correction has been performed as part of the data analysis, although as expected the differences between the matrix attenuation corrected and uncorrected ratios are small.

These U235 gamma peak ratios will be biased if there is a significant quantity of plutonium present in the same waste drum containing the uranium, which is common in Drigg waste drums. If Pu239 and Am241 are present, gamma peaks from Pu239 and Am241 may interfere with the U235 peaks identified in Table 2. The extent of the bias is dependent on the relative amounts of Pu239, Am241 and U235 and the "quality" of the gamma spectrum. There may also be other radionuclides present in the waste drum (e.g. Ra226) that could bias the U235 gamma peak ratios. The gamma spectrum must therefore be examined by an expert physicist to confirm the absence of interfering nuclides before relying on this technique. The fact that there are three ratios is therefore advantageous as the bias may affect only one of the peaks, with the other two remaining unbiased.

It is often possible to correct for interference to the U235 gamma peaks, using an appropriate reference peak from the interfering nuclide, and correcting for differences in branching ratio, matrix attenuation and detector efficiency. For example, the interference to the U235 143.8keV gamma peak from the Pu239 144.2keV gamma peak can be corrected by reference to the 129.3keV gamma peak, also from Pu239.

The DPA analysis is applied as follows to determine whether the uranium present in a waste drum exhibits significant self-shielding:

1. Identify the segments containing significant U235, and perform (600s) passive gamma and (300s) transmission measurements with the HRGS detector centred at each segment. Analyse the spectra using commercial software to obtain the net peak areas of the gamma peaks of interest.

2. Visually inspect the raw gamma spectra to determine whether the presence of interfering gamma peaks is likely to have significantly biased the determination of any of the net peak areas used in the calculation of the U235 peak ratios.

3. Calculate transmission corrected 143/186keV, 163/186keV and 205/186keV U235 peak ratios and associated statistical uncertainties. If necessary and possible, correct for the presence of any interference peaks identified in the previous step.

4. Compare the transmission corrected U235 peak ratios and uncertainties against the values expected for infinitely dilute uranium to determine whether the uranium is dilute. Note that the value plus two sigma confidence error must be less than the upper limit, and likewise, the value minus two sigma confidence error must be greater than the lower limit. The following ranges are considered to indicate dilute:

   $$0.14 < \text{143/186keV ratio} < 0.24$$
   $$0.07 < \text{163/186keV ratio} < 0.11$$
   $$0.07 < \text{205/186keV ratio} < 0.10$$

   Generally, all three ratios must be consistent with dilute uranium to pass, however, in cases where one of the three ratios is showing an inconsistency due to interference, then dilute uranium may be assigned.

To test the DPA analysis, measurements of a uranium calibration source known to exhibit significant self-shielding were performed. The source had a U235 mass of 18.6g in the form of U_3O_8 powder, with a calculated self-attenuation factor of 0.1041 (indicating that only approximately 10% of the 186keV gamma-rays penetrate to the outside of the source). The following peak ratios were obtained when the source was measured in an empty 200 litre drum.

143/186keV	163/186keV	205/186keV
0.107 ± 0.005	0.063 ± 0.004	0.106 ± 0.003

Table 3: U235 peak ratios for a self-shielding calibration source (1σ errors).

The ratios in Table 3 are consistent with a source that has significant self-shielding and are clearly outside the expected range for dilute uranium, demonstrating that the DPA technique can be used to identify uranium that exhibits significant self-shielding.

3.2. Self Induced X-Ray Fluorescence (SIXRF) Analysis

Fluorescence X-Rays can be stimulated by any alpha, beta and gamma ray emissions with energies greater than that of the X-Rays. The fluorescence X-Rays for uranium are listed in Table 4 below. In uranium, the gamma emissions are of low energy and intensity and it is the stimulation from alpha decay, and to a lesser extent beta decay, which predominate.

X-Ray	Energy, keV	Intensity, %
U $K\alpha_2$	94.66	61.9
U $K\alpha_1$	98.44	100.0
U $K\beta_3$	110.43	11.6
U $K\beta_1$	111.31	22.0
U $K\beta_2$	114.34, 114.57	12.3

Table 4: Fluorescence K X-Ray emissions from uranium [3].

In concentrated lumps of uranium, the alpha particle interactions cause ionisations, resulting in fluorescence X-Rays. In dilute uranium, the alpha particle interactions between uranium atoms are reduced and the fluorescence X-Rays are significantly smaller. This difference has therefore been used to identify whether the uranium present is in the form of a concentrated lump or dilute material.

The difference in the spectral shape with composition of the uranium can be clearly seen in Figures 2 and 3. For the purpose of comparison, the vertical scales have been adjusted to give approximately the same peak height for the U235 gamma emission at 143keV in both Figures. The spectrum in Figure 2 is from uranium in the form of bulk U_3O_8 powder of nominal 2% U235 enrichment, and shows intense photopeaks at 94.7 keV and 98.4 keV from the U Kα fluorescent X-rays. The spectrum in Figure 3 has the same nominal 2% enrichment; however, the uranium is in the form of uranyl nitrate solution. Since the uranium is more dispersed, there is reduced probability of self-induced fluorescence from alpha decays, hence a marked reduction in the intensity of the U Kα photopeaks.

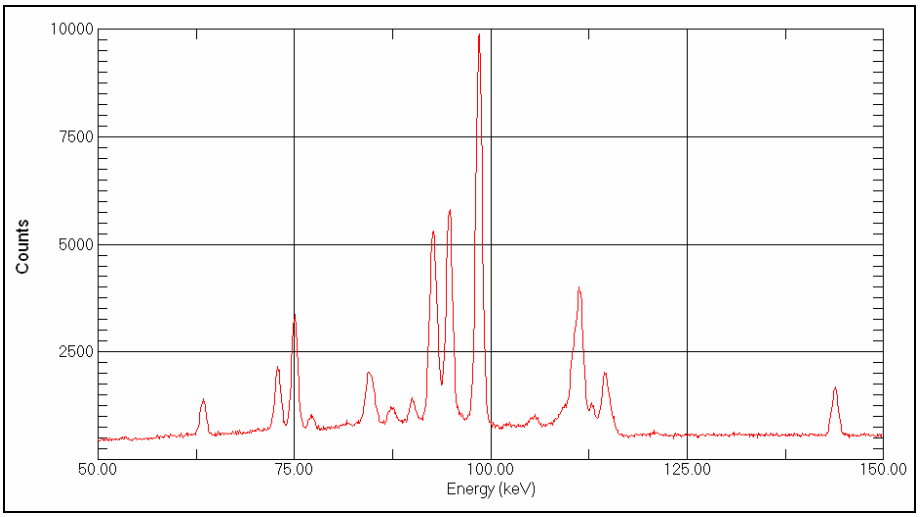

Figure 2: HRGS spectrum for U_3O_8 powder (2% U235)

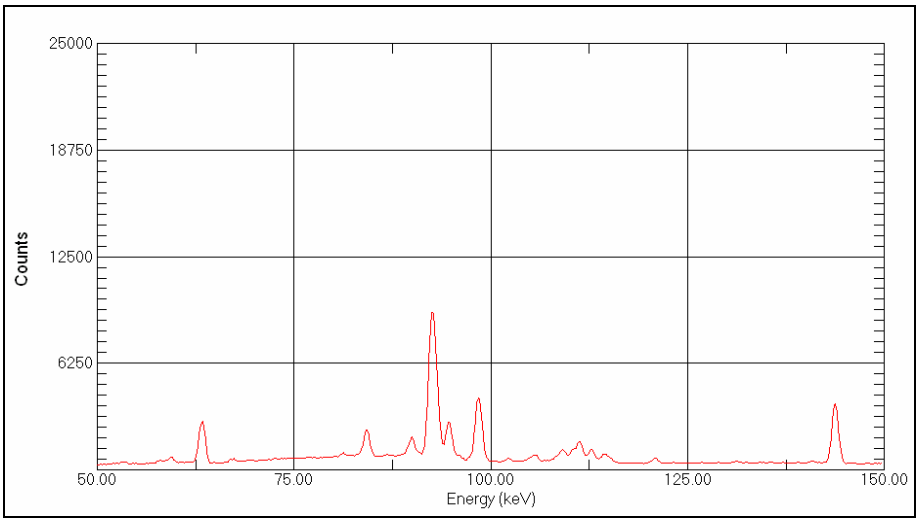

Figure 3: HRGS spectrum for 50gU/litre uranyl nitrate solution (2% U235)

The relative alpha emission rates for varying uranium enrichments are shown in Table 5. At low enrichments the total alpha emission rate is dominated by U238, however, it is soon dominated by U234 as the %U235 enrichment increases. Since the U234 content increases in proportion with the %U235 enrichment, the total alpha emission rate, and hence the intensity of the U Kα photopeaks, can be predicted for a known form of uranium, given a knowledge of the U235 and U238 content.

Isotope	Depleted U		Natural U		Low Enriched U	
	Mass %	Alpha Activity, Bq/gU	Mass %	Alpha Activity, Bq/gU	Mass %	Alpha Activity, Bq/gU
U234	0.002	4601	0.01	23007	0.021	48315
U235	0.301	241	0.713	570	2.014	1611
U238	99.7	12402	99.28	12350	97.93	12182
Total alpha activity, Bq/gU		17244		35927		62108

Table 5: Alpha activities for uranium with different enrichments.

To apply the SIXRF analysis technique, a relationship must be established between the U235 and U238 content, from the gamma spectrum, and the intensity of the U $K\alpha_1$ photopeak at 98.44 keV, for a reference material (e.g. bulk uranium oxide powders). Once this relationship is determined for the reference material, it can be applied to any measured gamma spectrum from an unknown form of uranium, to predict the intensity of the U $K\alpha_1$ photopeak. The actual measured intensity of the U $K\alpha_1$ photopeak is then ratioed to the predicted intensity for the reference material to determine the form of the unknown sample. If the unknown sample is of the same form as the reference material then a ratio of unity is to be expected, with values of less than unity indicating that the uranium in the sample is more dispersed. Similarly, a ratio higher than unity would indicate that the unknown sample is a denser form of uranium than the reference material.

To facilitate this analysis, count rates related to U235, U238 and the U Kα photopeaks need to be extracted from the spectrum. There are several possible choices of peaks representative of these quantities but, for simplicity, the peak areas are calculated using summation of Regions of Interest (ROI's) and no correction is made to the peak area counts for differences in detection efficiency. Therefore, to minimise potential errors, the ROI's are selected to be close in energy. The regions selected are shown in Figure 4 below, with the main emissions in these regions listed in Table 6.

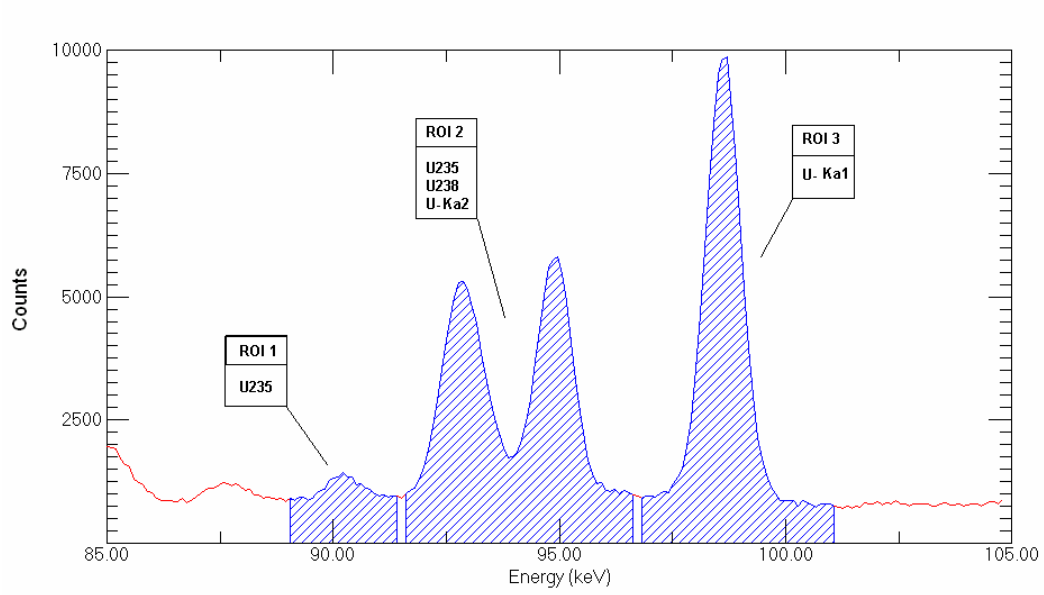

Figure 4: HRGS spectrum for reference U_3O_8 powder (2% U235) showing ROI's used for analysis

ROI	Parent	Origin	Energy [2] (keV)	Intensity [2] (%)	Total Intensity (%)
1 (88.6 keV to 91.1 keV)	U235	Th $K\alpha_2$ Th231	89.953 89.900	3.758 0.940	4.698
2 (91.2 keV to 96.7 keV)	U235	Pa $K\alpha_2$ Th $K\alpha_1$ Pa $K\alpha_1$	92.290 93.350 95.868	0.399 6.131 0.647	7.177
	U238	Th234 Th234	92.368 92.800	2.721 2.689	5.410
	U fluorescence	U $K\alpha_2$	94.660	61.900	61.900
3 (96.9 keV to 100.4 keV)	U fluorescence	U $K\alpha_1$	98.440	100.00	100.000

Table 6: Gamma and X-Ray Emissions in the analysis ROI's.

ROI 2 is a complex with contributions from U235, U238 and the U $K\alpha_2$ X-Ray. The U238 counts can be extracted by stripping out the U235 and U $K\alpha_2$ counts from this region. To facilitate this requires calibration using samples of a reference material, i.e. samples of the same form (e.g. bulk oxide powder) but with a range of enrichments. A fluorescence ratio (F_{Ratio}) can then be calculated for the unknown sample, to determine the amount of fluorescence relative to the reference material.

Note that if plutonium and uranium are present in the same waste drum, then the alpha emission will be significantly higher and fluorescence X-Rays will be present from both plutonium and uranium. This is easily identifiable due to the presence of characteristic plutonium and uranium X-Rays and gamma rays. For simplicity, the SIXRF analysis described is not applied when interference due to the enhanced fluorescence of the uranium from the plutonium is either confirmed or suspected. However, extensions to this technique, which take into account these effects, are possible.

4. Results

The 12 drums identified in Table 1 were repacked and the derivative drums remeasured using the Drigg Drum Monitors. Any derivative drums containing significant U235 mass (nominally >50g) were then subjected to additional measurements to obtain high quality gamma spectra for the DPA and SIXRF analyses. The results from two of these drums are discussed in detail in Sections 4.1 and 4.2, and an overall summary of the analysis results for all 12 drums is presented in Section 4.3.

4.1. Drum 1

When Drum 1 was opened, it was found to contain a smaller inner drum with 15 waste packets inside. These packets were removed, inspected and placed within 11 new drums. A summary of the inspection of the packets removed from Drum 1 is provided in Table 7. Two photos taken during the repacking of this drum are shown in Figure 5.

Repacked Drum Packet Number	Packet Weight, kg	Approximate Dimensions, mm	Description of Contents
Packet A	5	not estimated	• Black material, rubber
Packet B	3	not estimated	• White powder
Packet C	5	not estimated	• Black material, glovebox gloves
Packet D	3	200 × 100	• Glovebox gloves
Packet E	3	200 × 100	• Glovebox gloves
Packet F	2.5	200 × 100	• White powder
Packet G	3	200 × 100	• Glovebox glove
Packet H & Packet I	2 2.5	100 × 100 150 × 150	• White powder • Black powder
Packet J & Packet K	2.5 2.5	150 × 150 200 × 100	• Black powder • Glovebox glove
Packet L & Packet M	3 2	200 × 100 200 × 50	• Black powder • Wrapped PVC
Original Drum Containing Packet N & Packet O	2.5 2	150 × 50 150 × 150	Original drum 1 • Black powder • Powder

Table 7: Summary of waste packets removed from Drum 1

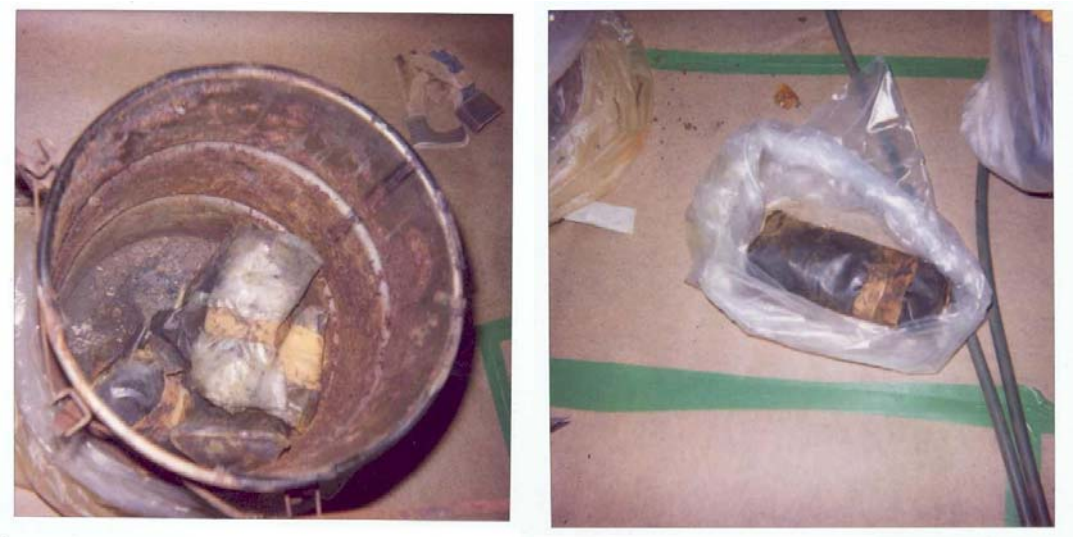

Figure 5: Example photos taken during repacking of Drum 1 (showing Inner Drum and Packet C)

The physical descriptions of these packets indicate the presence of either white or black powder in a number of the packets. There are also several glovebox gloves, and it is suspected that the waste was generated during the clean up of a glovebox. The fissile mass assay results for the repacked drums are shown in Table 8 together with the DPA analysis results. In this case, all 11 derivative drums contained a significant U235 mass and were subjected to further analysis. The assay results for the original drum before repacking (from Table 1) have also been included for ease of reference.

Repacked Drum Packet Number	Drum Monitor Assay Results			DPA U235 Peak Ratio Results ± 2σ statistical uncertainty		
	Total Fissile, g	Total Pu, g	U235, g	143/186keV	163/186keV	205/186keV
Packet A	104.05	0.441	103.61	0.194 ± 0.013	0.094 ± 0.007	0.085 ± 0.005
Packet B	37.7	0.529	37.171	0.195 ± 0.013	0.089 ± 0.009	0.090 ± 0.008
Packet C	76.063	0.458	75.606	0.189 ± 0.010	0.099 ± 0.007	0.088 ± 0.005
Packet D	78.043	0.424	77.619	0.180 ± 0.008	0.091 ± 0.006	0.089 ± 0.005
Packet E	58.808	0.467	58.341	0.192 ± 0.012	0.090 ± 0.007	0.087 ± 0.006
Packet F	48.995	0.396	48.599	0.192 ± 0.013	0.081 ± 0.008	0.084 ± 0.007
Packet G	98.896	0.423	98.473	0.201 ± 0.009	0.091 ± 0.005	0.086 ± 0.005
Packets H & I	111.857	0.447	111.41	0.198 ± 0.012	0.091 ± 0.006	0.088 ± 0.003
Packets J & K	71.282	0.522	70.76	0.186 ± 0.011	0.091 ± 0.007	0.089 ± 0.006
Packets L & M	81.643	0.443	81.2	0.178 ± 0.008	0.088 ± 0.006	0.088 ± 0.005
Original Drum containing Packets N & O	119.853	0.608	119.245	0.170 ± 0.008	0.090 ± 0.006	0.091 ± 0.005
Sum Totals	**887.19**	**5.158**	**882.034**			
Original Assay	**1564.06**	**0.74**	**1563.33**			

Table 8: Fissile mass assay and DPA analysis results for repacked drums from Drum 1

Examination of the fissile mass results indicates that the sum total U235 and Pu mass results for the individual packets differ significantly from the original assay (before repacking). The reduction in the U235 mass can be explained by the reduction in waste density and centering of the waste packets during the repack, giving a more accurate assay (see (iii) in Section 3.1). The apparent increase in the Pu mass is because each drum has been assigned the detection limit, as there is no Pu present.

The DPA analysis results in Table 8 show good consistency, indicating that the uranium in each packet is equivalent in terms of self-shielding properties. All three U235 peak ratios are clearly consistent with the ranges for infinitely dilute (i.e. non-self-shielding) uranium.

Figure 6 shows the fluorescence X-Ray region of the gamma spectrum for one of the derivative drums from Drum 1. Although evident, the uranium fluorescence X-Rays are small relative to the U235 143keV and Th231 (U235) 84.2keV gamma peaks either side. Using bulk uranium oxide powder as the reference material, a fluorescence ratio (F_{Ratio}) of 0.08 ± 0.04 (1 sigma error) is calculated, indicating that the uranium is significantly less concentrated than the reference material.

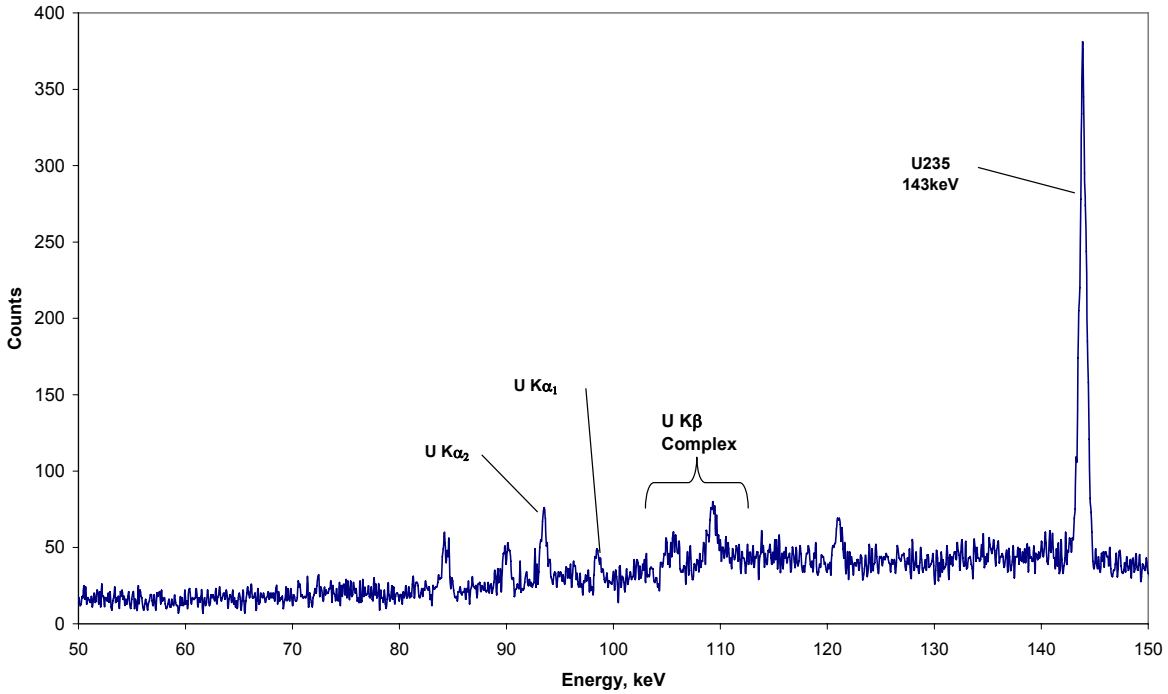

Figure 6: HRGS spectrum for the original drum containing waste packets N & O after repacking.

In summary, there is very strong evidence to indicate that Drum 1 does not contain lumps of uranium. The visual inspection of the packets contained within this drum indicated the presence of powder, and both the DPA and SIXRF analyses provide very convincing evidence for dilute uranium.

4.2. Drum 9

When Drum 9 was repacked, it was found to contain 33 separate waste packets. These packets were removed, inspected and placed within 22 new drums. An example photo taken during the repacking of this drum is shown in Figure 7. A summary of the waste packet inspection has not been included due to the large number of packets; however, the majority of the physical descriptions were of metallic items such as billets, tins, pellets and residues.

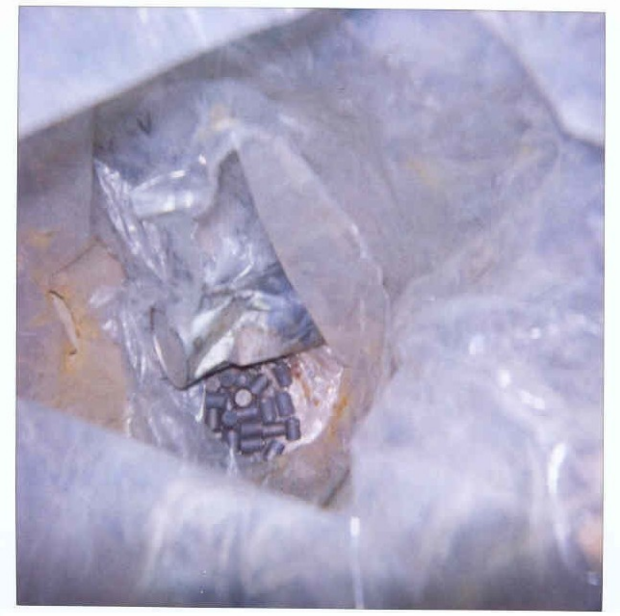

Figure 7: Example photo taken during repacking of Drum 9 (Packet AG)

The physical descriptions of these packets suggest that the uranium present within these drums may be exhibit significant self-shielding properties due to the presence of pellets, billets and residues. The fissile mass assay results for the repacked drums are shown in Table 9 together with the DPA analysis results. Seven of the packets either contained a significant U235 mass (i.e. >50g), or the physical description and photographs gave reason to believe there may be lumps of uranium present, and were subjected to further analysis. Note that several measurements of these repacked drums failed the transmission measurement due to "High Density" of the waste and no results were reported. These drums will be subjected to special investigation later. Because of the incomplete set of results, the individual packet results have not been compared against the original drum results (before repack).

Repacked Drum Packet Number	Drum Monitor Assay Results			DPA U235 Peak Ratio Results ± 2σ statistical uncertainty		
	Total Fissile, g	Total Pu, g	U235, g	143/186keV	163/186keV	205/186keV
Packets A & B	Failed due to "High Density"			Not measured		
Packets C & D	78.064	2.538	76.075	0.100 ± 0.011	0.072 ± 0.009	0.104 ± 0.010
Packets E & F	23.723	1.345	22.901	Not measured		
Packets G, H & I	106.625	0.452	106.173	0.130 ± 0.025	0.059 ± 0.011	0.102 ± 0.012
Packets J & K	130.806	1.453	129.807	0.125 ± 0.016	0.067 ± 0.011	0.094 ± 0.011
Packets L & M	33.007	0.445	32.563	Not measured		
Packets N & O	Failed due to "High Density"			Not measured		
Packets P & Q	9.989	0.406	9.583	Not measured		
Packets R & S	39.461	0.516	38.944	Not measured		
Packets T & U	70.425	1.763	69.11	0.113 ± 0.022	0.079 ± 0.018	0.096 ± 0.016
Packet V	2.634	0.653	1.981	Not measured		
Packet W	Failed due to "High Density"			Not measured		
Packet X	12.950	1.065	12.265	Not measured		
Packet Y	27.931	6.529	21.402	Not measured		
Packet Z	44.582	3.200	41.951	Not measured		
Packet AA	2.709	0.548	2.161	Not measured		
Packet AB	8.599	0.460	8.138	Not measured		
Packet AC	Failed due to "High Density"			Not measured		
Packet AD	69.571	3.208	67.043	0.126 ± 0.016	0.065 ± 0.012	0.085 ± 0.013
Packet AE	10.176	0.514	9.661	Not measured		
Packet AF	20.636	2.590	18.611	0.117 ± 0.030	0.083 ± 0.028	0.084 ± 0.024
Packet AG	46.934	4.836	42.723	0.120 ± 0.019	0.059 ± 0.014	0.100 ± 0.015

Table 9: Fissile mass assay and DPA analysis results for repacked drums from Drum 9.

The DPA results in Table 9 again show good consistency, indicating that the uranium in each packet is similar in terms of self-shielding properties. The U235 peak ratios are inconsistent with dilute uranium and are very similar to the values obtained with the uranium calibration source reported in Table 3, indicating that the uranium present exhibits significant self-shielding.

Isotopic analysis of these waste packets was performed using MGA-U [4], which indicated that the uranium was indicates that the uranium was of very low enrichment (~1% U235). The drum containing waste packets J & K had a reported U235 mass of ≈130g. Using the measured enrichment of 1%, this equates to ≈13kg of uranium. However, the combined weight of these packets was only 6kg indicating that the U235 mass must have been significantly overestimated. In this case, the overestimation is likely to be due to segment overlap (as in Section 3.1) because the uranium is known to exhibit significant self-shielding.

In summary, there is very strong evidence to indicate that Drum 9 does contain lumps of uranium and therefore the result obtained from the B746 Drum Monitor is assumed to be accurate in this case. Isotopic analysis indicates that the uranium originating from Drum 9 is natural or of very low enrichment (~1% U235).

Figure 8 shows the fluorescence X-Ray region of the gamma spectrum for one of the derivative drums from Drum 9. In this case the uranium fluorescence X-Rays are very large relative to the U235 143keV and Th231 84.2keV gamma peaks either side. Using bulk uranium oxide powder as the reference material, a fluorescence ratio (F_{Ratio}) of 0.54 ± 0.07 (1 sigma error) is calculated, indicating that the uranium is of similar concentration to the reference material.

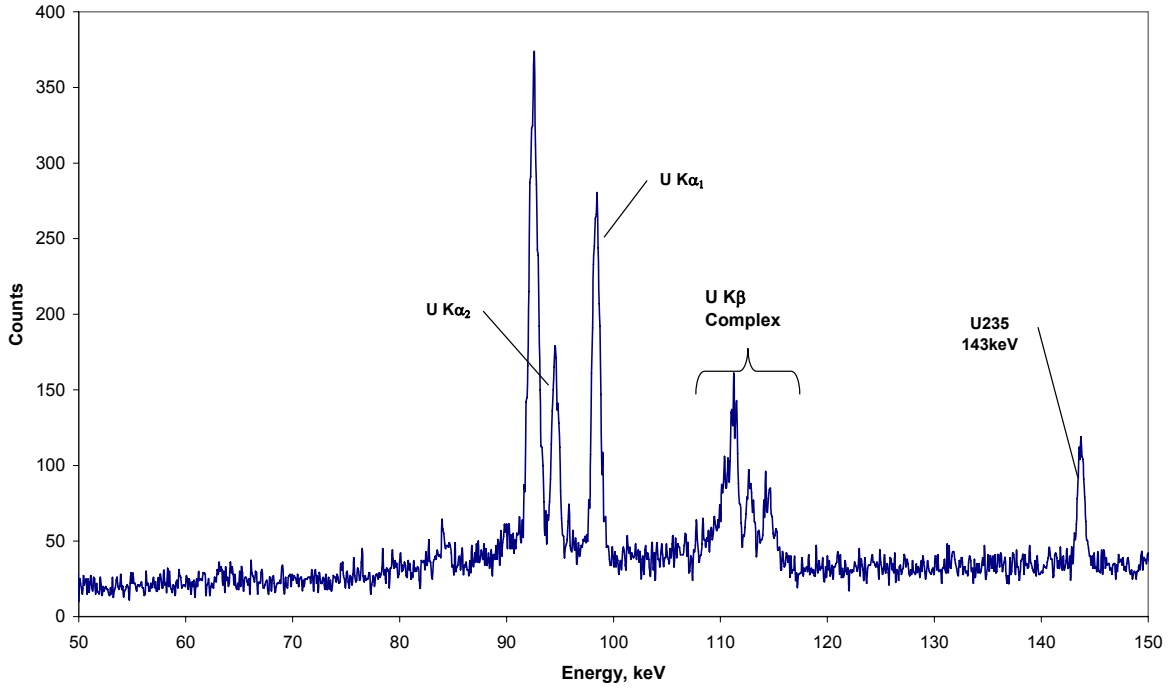

Figure 8: HRGS spectrum for waste packets G, H & I removed from Drum 9.

In summary, there is very strong evidence to indicate that Drum 9 contains lumps of uranium. The visual inspection of the packets contained within this drum indicated the presence of pellets, billets and residues, and both the DPA and SIXRF analyses provide very convincing evidence that the uranium exhibits significant self-shielding.

4.3. Summary Results for All Drums

The DPA analysis results for the 12 repacked drums are summarised graphically in Figures 9 to 11. For clarity, only one result has been shown for each drum, despite several derivative drums from each original drum being analysed in each case. This is justified, however, since the DPA results obtained for each derivative drum were always consistent with other derivative drums from the same original drum.

The results indicate that from the original sample of 12 drums, 2 drums (i.e. Drums 8 and 9) were found to contain uranium with significant self-shielding. Although not shown here, the SIXRF analysis results were consistent with the DPA results, and indicated that only Drums 8 and 9 contained uranium lumps.

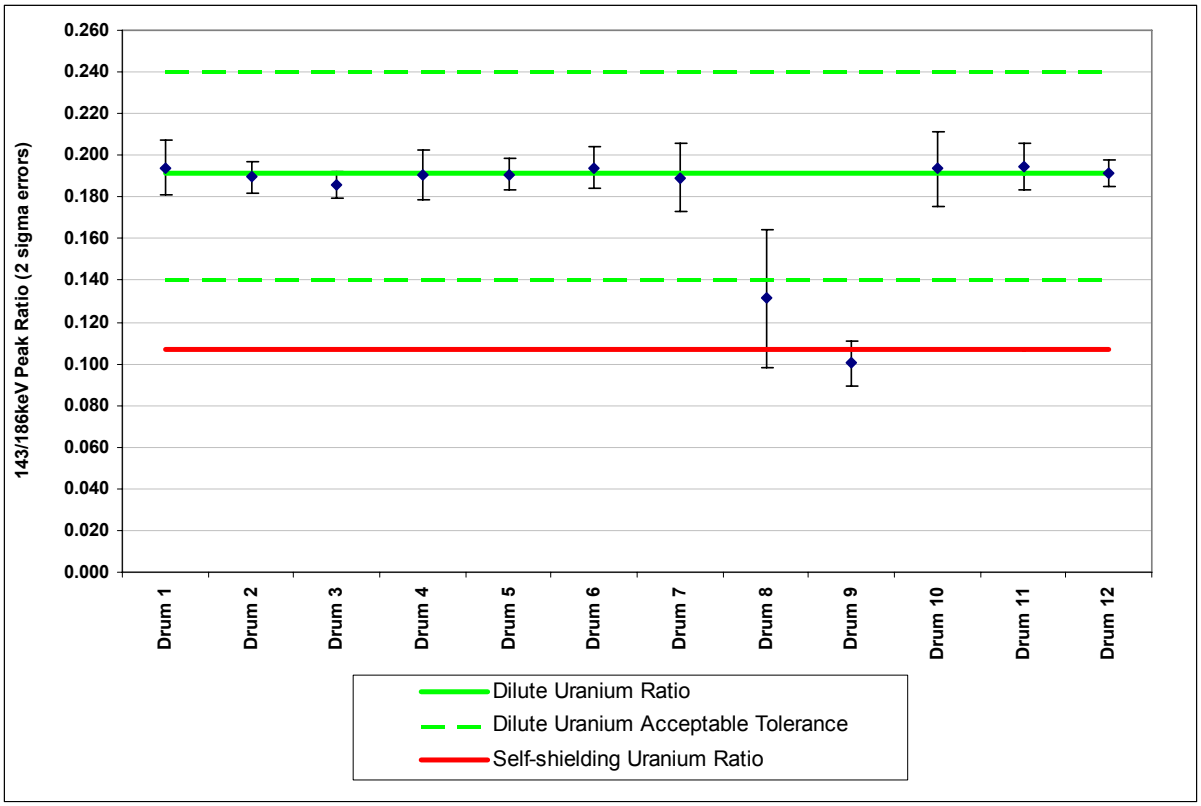

Figure 9: DPA Analysis U235 143/186keV peak ratio results for 12 repacked drums.

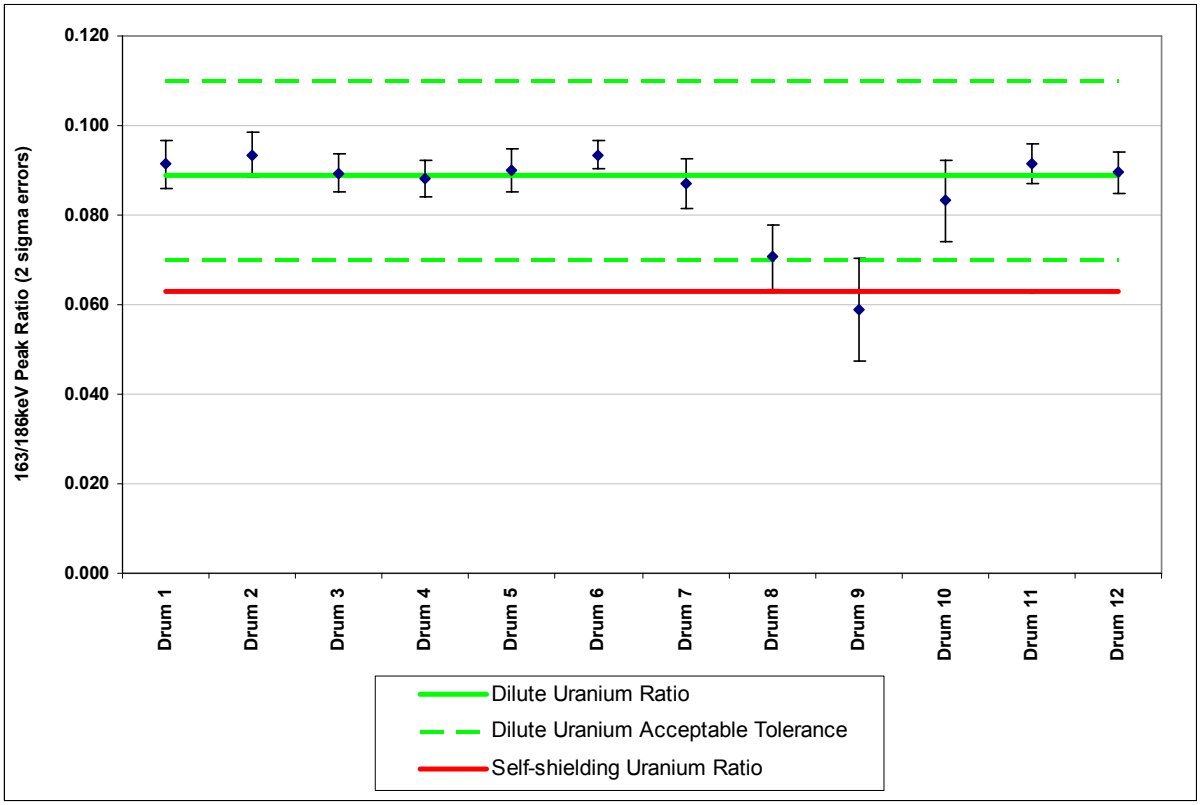

Figure 10: DPA Analysis U235 163/186keV peak ratio results for 12 repacked drums.

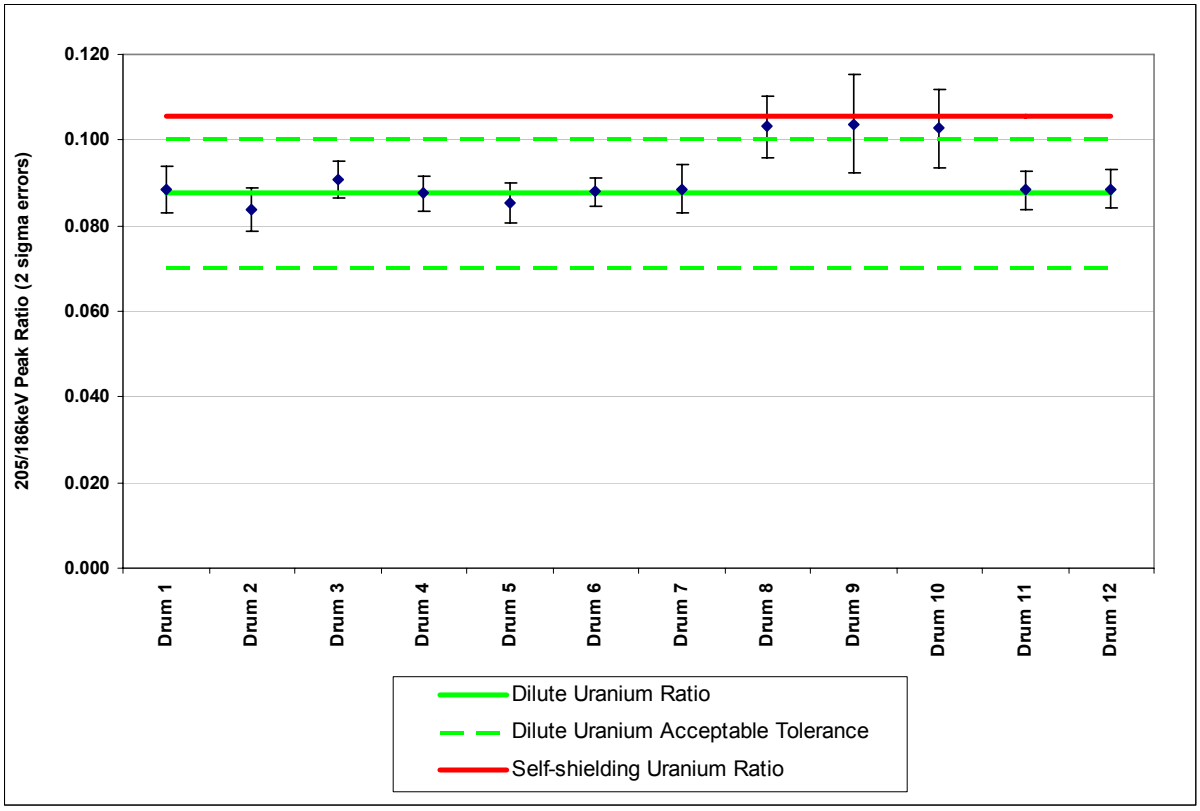

Figure 11: DPA Analysis U235 205/186keV peak ratio results for 12 repacked drums.

5. Conclusions

Although the discovery of 2 drums containing lumps of uranium from the sample of 12 was unexpected, the experience gained from the measurements of the repacked drums has demonstrated the validity and usefulness of both the DPA and SIXRF techniques for identifying waste drums that contain dilute or lumpy uranium. For drums found to contain dilute uranium, the U235 assay results reported by the Drigg Drum Monitor can be reduced by a factor of 8.33 to remove the uranium self-shielding factor assumed in the calibration. This will significantly improve the accuracy of the U235 assay results.

In this work, the measurements and analysis were performed on drums that had already been repacked; however, the same methodology has been successfully applied to a large number of general waste drums. This has significantly reduced the number of drums that would otherwise require physical repacking to comply with the fissile mass limits for safe transport and storage.

The techniques do have limitations when applied to Drigg wastes, however, mainly due to the potential for interference from plutonium and Am241. As a consequence, the gamma spectra must be inspected by an expert physicist to confirm the validity of the results, and there are currently no plans to implement the analysis automatically within the drum monitors.

The results obtained for different packets removed from the same original drum show excellent consistency with respect to self-shielding properties, and there was no evidence to suggest that mixtures of both dilute and lumpy uranium were present in any of the drums. The DPA technique assumes that all uranium within the assayed drum has the same properties with respect to self-shielding. If there is a mixture of dilute and lumpy material, then the dilute material may dominate and the lump may not be identified, due to the higher signal per gram of the dilute material. It should also be noted that, since the analysis is applied to each high U235 segment, the mixtures would need to be present in the same segment for there to be bias, which further reduces the likelihood.

In this application, the DPA and SIXRF techniques are used to provide conclusive evidence of whether the uranium is dilute, to support the removal of the assumed self-shielding in the uranium calibration. The same techniques could easily be adapted to provide a uranium lump correction factor, for cases when lumps of uranium are detected, although this would require some additional calibration work. Both techniques could also be applied to plutonium assay to improve the accuracy of quantitative gamma based analysis when lumps of plutonium are present.

6. References

[1] Briermeister, J F; *MCNP – A General Monte Carlo N-Particle Transport Code, Version* 4B; Los Alamos National Laboratory; LA-12625-M, Version 4B; 1997.

[2] *JEF-PC*; OECD / NEA Data Bank; 1994; JEF 2.2 decay database.

[3] Reilly, D, Ensslin, N, Hastings Smith H, and Kreiner, S; *Passive Nondestructive Assay of Nuclear Materials*; NUREG/CR-5550, LA-UR-90-732; ISBN 0-16-032724-5; 1991; p161-164, p316.

[4] *Multi-Group Analysis MGA++, The U235 Program, Uranium Isotopic Abundance by Gamma-Ray Spectroscopy, U235View, Part 2 of MGA-B32, Software User's Manual*; ORTEC Part Number 779950; 2002.

Safeguards in Geological Repositories for Spent Fuel

Site Definition and Design Information Verification

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Abstract

The international safeguards community adopted the issue of final disposal of spent fuel in 1988, when an IAEA Advisory Group recommended, as long as the world community kept IAEA Safeguards in force anywhere, not to terminate safeguards on spent nuclear fuel. As of that time, a number of concerned member states pursued the safeguards issue realising that geological disposal of spent fuel would bring forward hitherto unknown safeguards challenges. From 1999 till 2005, the Belgian, Canadian, Czech, Euratom, Finnish, German, Hungarian, Swedish, and US support programmes to the International Atomic Energy Agency (IAEA) have cooperated in a joint task. Once a year, the international experts have been convening to discuss the emerging issues. The last experts group meeting took place in Germany in June 2004. The theme of the meeting was "site definition and design information in relation to baseline information for a spent fuel repository with particular application to active repository projects". In this connection, the potential of various technical methods was discussed to support safeguards in geological repositories. The experts also addressed the implications of the Additional Protocol to reduce monitoring and physical inventory taking. The paper gives an overview of the outcome of the June 2004 experts group meeting.

Keywords: geological repository, final disposal, site definition, design information, geophysical methods

1. Introduction

Several countries and the Euratom Safeguards Authority support the International Atomic Energy Agency (IAEA) in the development of safeguards for geological repositories. Since 1999, international experts have been convening once a year to identify and discuss relevant issues. The last experts group meeting was organised by the German Federal Office for Radiation Protection at its headquarters in Salzgitter/Germany in June 2004. The theme of the meeting was "site definition and design information in relation to baseline information for a spent fuel repository with particular application to active repository projects". In this connection, the potential of various technical methods was discussed to support IAEA safeguards in geological repositories, among them satellite imagery, environmental monitoring methods, ground penetrating radar, and other geoelectrical measurement methods. The experts also addressed the implications of the Additional Protocol to reduce verification needs. This paper gives an overview of the outcome of the experts group meeting [1].

[1] Retired from the Federal Office for Radiation Protection, D-38201 Salzgitter, Germany

2. Status of Repository Projects

Belgium has an underground research facility at the Mol Nuclear Research Centre. The facility is situated at a depth of 230 m in the Boom Clay. The research objective is to study the suitability of clay for the final disposal of nuclear waste and to develop a technical repository concept.

In Finland, the final disposal of spent nuclear fuel in a geological repository at Olkiluoto was accepted by the local municipality, state authorities, and finally endorsed by the Finnish Parliament. In 2004, the excavation of the tunnel system for bedrock characterisation started. The bedrock is mainly composed of metasedimentary migmatic mica gneisses which are cut by granodiorite, coarse grained granites and granitic pegmatites [2]. The as-built information of the geological repository will be generated using conventional technology from underground surveying. Hydrogeological site characterisation and monitoring is continuing.

Germany has a licensed nuclear repository for non-heat-generating radioactive waste in an iron ore formation awaiting court decision on its going into operation. At another site, characterisation is going on for final disposal of all sorts of radioactive waste including high level waste and spent nuclear fuel in rock salt. In October 2000, the German government stopped the characterisation activities for up to 10 years. The current policy is to pursue only one repository for all types of waste and to consider and evaluate final disposal also in other geological formations.

In Sweden, two possible nuclear sites had been selected for a nuclear repository: Forsmark and Oskarshamn. Final site selection is planned to take place in 2007 and license application in 2008. The repository medium is crystalline bedrock, similar to the Finnish medium.

In the USA, the plan is to operate a nuclear repository at Yucca Mountain (YM). The repository is being built in welded tuff above the groundwater table. YM would be licensed for spent fuel from commercial power reactors, research reactors, vitrified high-level radioactive waste, and military waste. Final disposal casks would be laid in tunnels. While it is planned to start operation in 2010, no decision has been taken on the schedule for backfilling and on whether YM would become eligible for IAEA safeguards.

3. Safeguards Objectives

In comparison to existing nuclear facilities geological repositories will pose novel requirements for safeguards. Due to their structural characteristics and due to the international consensus not to terminate IAEA Safeguards on spent fuel, as long as safeguards will be applied elsewhere [3], it is necessary to distinguish between three facility phases, i.e., (1) pre-operational, (2) operational, and (3) post-closure. The pre-operational phase begins when a state declares a location to become the site of a nuclear repository and construction of the repository begins. At that point in time, an underground structure may already exist as a result of site characterisation. The operational phase begins when the first nuclear material package is received at the facility. The post-closure phase begins when the repository is filled, closed and decommissioned. The time scale is unprecedented. Underground site characterisation takes many years without the IAEA being involved. The construction phase may last for 5 years, and the operational phase may last for 70 years.

From a safeguards point of view, it will be difficult to define the physical boundaries of a repository and to select adequate measures for the verification of the repository design. Misuse of the repository could be prepared during site characterization and implemented during the operational as well as post-closure phases.

Therefore, the objectives of the IAEA Safeguards related to geological repositories for spent nuclear fuel are to provide a high level of assurance that

- the spent fuel declared to be transferred into a repository is actually present there,
- undeclared removal of nuclear material is detected,
- the repository is actually built and operated as declared,
- undeclared excavations and operations are detected.

4. Site Definition

A site declaration is required under the Additional Protocol (AP) [4]. The USA provides site declarations as part of the design information under their existing Voluntary Offer Safeguards Agreement. The experts agreed to consider the site declaration issue in accordance with the AP. To the extent necessary for the discussions the site is defined as follows:

"A site means that area delimited by (State) in the relevant design information for a facility [...] where nuclear material is customarily used [...]. It shall also include all installations, co-located with the facility or location, for the provision or use of essential services, including: hot cells for processing irradiated materials not containing nuclear material; installations for the treatment, storage and disposal of waste; [...]."

Normally, the security fence determines the site boundaries of a nuclear facility. However, in the case of a geological repository this simple approach may be insufficient, as a repository has a surface part and an underground part.

For some experts the question of how to define the site boundaries was closely linked to the question of how to detect undeclared access to the disposed of nuclear material. One concern was that the facility area enclosed by the security fence may be smaller than the projection of the underground structures to the surface. It was proposed to extend the site boundaries accordingly, in order to permit inspector's access to the area directly above the underground structures. Thus, the inspector could make maximum use of visual inspections and, if possible, apply geophysical monitoring, in order to detect undeclared activities in the underground. The possibility of tunneling from a neighbouring mine, if it exists, was also mentioned. The following factors were discussed without reaching a conclusion:

- To define a distance from the declared excavations at which undeclared access could be attempted would be arbitrary and of no substantial value.
- The velocity at which tunnels could be excavated would depend on the geological medium and may not be representative during site characterisation. Therefore, assuming an order of magnitude of 1 km per year may be of no value.
- The time span within which a disposal package could be retrieved from a closed repository was found to be more than one but less than 2 years [5].
- With respect to post-closure safeguards of the repository it was stated that the conditions for the IAEA to carry out inspections and to envisage geophysical monitoring from above a closed repository would be site-specific. The land may be government-owned or public with the possibility to agree upon monitoring. Private ownership would imply more restrictions.
- Under the AP the IAEA would be able to request Complementary Access to any location, also outside the repository site, within a given time span, once a state was unable to resolve an inconsistency, the AP would not provide the possibility to install geophysical monitoring systems.
- In a preceding experts group meeting the possibility for the IAEA to use facility operator's monitoring systems was positively analysed. Also, the IAEA might be provided safety-relevant data acquired in connection with site characterisation and thereafter. However, these approaches have to bear in mind that the IAEA required authenticated data. Alternatively, the IAEA might also be granted the right to install its own geophysical system.

In trying to reach an understanding of and, possibly, agreement on the site boundaries the experts discussed several proposals:

- To consider only the surface area within the security fence, while there may be more than one such area.
- With respect to the underground part, to consider only the excavated parts which are accessible.
- To consider three-dimensional boundaries, i.e., "block of rock", to comprise underground part, surface part and buffer zone; i.e., to define a virtual repository wall defined by geographical coordinates.
- The site boundary should be related to the state making a statement on whether there was a distance from the repository within which it would be forbidden to perform other mining operations.
- While the site should be defined exclusively by the above-ground facility, there should be a containment concept with respect to the underground facility.
- To consider all the state-owned land above the repository as the site.

A majority of experts endorsed the proposal to define the repository site by the security-fenced area above ground and the walls of the excavations below ground including all backfilled emplacement drifts and tunnels. However, no principal agreement was reached on the resolution of the repository site issue.

5. Design Information, Design Information Verification, and Baseline Information

Upon declaration of a new facility the state is to provide sufficient design information (DI). In Euratom terminology the basic technical characteristics are to be provided regarding the as-built structures of the facility as well as the operational processes. Based on this information the IAEA develops a facility-specific safeguards approach.

The experts considered the following features to be fundamental for a geological repository. Above ground there are facilities for shipping and receiving transport casks and for repackaging spent fuel into final disposal packages, shaft and ramp head-ends as well as buildings related to the administration, security, and maintenance. Underground structures are shafts, ramps, tunnels, drifts, and rooms.

The IAEA's mechanism to acquire DI from a state is to submit a design information questionnaire (DIQ). With respect to the surface part it was proposed that the state should provide the DI required for "Separate Storage Installations"-type facilities. For underground facilities there is no DIQ.

Underground parts of geological repositories appear to be much less transparent than hitherto existing above-ground nuclear facilities. Therefore, it has been suggested to define, complementary to DI, baseline information intended to support the IAEA in fulfilling its mission especially in terms of the detection of undeclared nuclear activities. Baseline information may consist of site-specific data collected for site characterization, but also during construction of the repository. In essence, it could relate to and comprise the following:

- Site geology, hydrology, geophysics
- imagery of undisturbed site including elevation model
- repository concept
- locations of all boreholes.

Further detailed baseline information could be the following, while, in Europe, there is no legal basis for providing such information:

- Conclusions on the integrity of the host rock;
- width of buffer zones and control areas established around the repository;
- acceptable locations and depths for installing geophysical sensors;
- identification of all underground activities within 10 km of the excavated repository not related to the repository;
- information about large structures within 5 km of the excavated repository which could be used for concealment, e.g., of undeclared shaft head-ends;
- identification of all significant acoustic or microseismic sources whose emissions are transmitted within the host rock;
- identification of measures used to strengthen or stabilize tunnel walls, excavation walls and ceilings;
- identification of radionuclides found in the repository environment.

For design information verification (DIV) satellite imagery and ground penetrating radar (GPR) might be feasible methods.

For the detection of undeclared activities radionuclide analysis, passive seismic and acoustic monitoring, and satellite imagery might be feasible methods.

Considering the above-mentioned "block of rock"-approach with the security-fenced surface area and the projection of the underground structures as defining the "safeguards containment of a geological repository", it would be very difficult for the IAEA, if not impossible, to verify the integrity of the host rock and to authenticate the data on which the construction license would be based, even if the

above-specified baseline information was made available. Concerning repository surroundings, the IAEA would have to rely on open source information used for state evaluation.

The use of synthetic aperture radar data from satellite-based sensors for semi-automatic site monitoring and high-resolution panchromatic data for change detection could serve both DIV and detection of undeclared activities. Pixel-based as well as object-oriented change detection methods using multi-spectral data and high-resolution panchromatic data from satellite-based sensors may also have a safeguards potential. Satellite imagery could help to confirm the completeness of DI and contribute to the data base of the baseline data.

The IAEA is lacking guidelines on how to use satellite imagery for geological repositories and an evaluation of the different satellite imagery techniques with regard to their capability to detect undeclared mining activities. The IAEA is particularly looking for radar and hyperspectral imagery, as well as for expertise, funding, and sensor platforms. The feasibility of satellite imagery has to be evaluated on a site-specific basis. In the post-closure phase, costs might be reduced, if satellite imagery would be used to verify state's compliance with several treaties, i.e., not only the NPT.

GPR and other geoelectrical measurement methods are being considered for DIV. Theoretical model investigations showed, however, that within salt, clay/claystone and crystalline rock there would be no general feasibility. The same holds true for acoustic and seismic monitoring methods. A major problem encountered in European countries is the occurrence not only of natural but also of industrial noise which may influence the application of these methods. Furthermore, there are legal problems due to private landownership, if measuring equipment would have to be installed in a wider area. The feasibility of GPR and other geoelectrical measurement methods for DIV would have to be studied on a case-by-case basis.

In plastic clay host rock there are indications that it is feasible to monitor the water pressure and the temperature to detect undeclared underground activities. In the post-closure phase, this type of monitoring could be continued, as deep boreholes for the sensors of the monitoring system would pose no safety problem in clay that was already saturated with water.

A laser range finder method similar to the one developed by the JRC Ispra ("3D Laser Range Scanner for DIV", LRSD) may be a possible technique for DIV, complemented by photographic imaging, if agreeable to the plant operator. The applicability of the LRSD for verifying the as-built underground repository would have to be studied on a case-by-case basis for the following reasons. Rock salt has the property of converging, tunnel dimensions are constantly changing and thus would be different when a re-measurement was taken. Similarly, in granite, there were stress movements which might yield false alarms in connection with the LRSD. It might easier to apply the LRSD method for the verification of closures/sealing installed at backfilled emplacement drifts.

The inspector would have to have means to define his location within the facility. The operator would have to provide drawings of the (as-planned and) as-built underground repository. Furthermore, the IAEA might be interested in authenticating data provided by the plant operator. Other factors to be considered and monitored might be the types of operator's equipment used below ground, the width of tunnels, and the possibilities for concealment of undeclared tunnels.

The applicability of radionuclide analysis, i.e., environmental monitoring, to detect undeclared nuclear activities in a geological repository strongly depends on the presence of natural isotopes. Therefore, it is not possible to make an a priori statement on the applicability of environmental monitoring in a geological repository. The feasibility would have to be studied on a case-by-case basis.

In order to limit both the safeguards costs and the burden on the facility operator, the IAEA might be able to use monitoring data acquired by the state and operator. This is only acceptable, if the data are authenticated, even though the large variety and redundancy of the data would make it very difficult for a state to succeed in consistently falsifying the data, in order to conceal undeclared mining activities.

The experts concluded that the IAEA needed recommendations on DI, DIV, and baseline information. Furthermore, the construction of a geological repository should be bound to the pre-condition of an Additional Protocol in force.

6. Further Implications of the Additional Protocol

The experts addressed the timing for delivering the first site declaration. It was noted that a repository project would be described in the state's 10-years nuclear plan required under the AP. This information would be provided in anticipation of the first declaration, i.e., before the start of the pre-operational phase. In addition, Member States Support Programmes to the IAEA could be used as a vehicle for early notice and cooperation.

Another AP-related issue was the IAEA's state evaluation. This should have a major influence on the IAEA's verification needs. As a result of a state evaluation, the IAEA should be able to come to a conclusion whether or not there is credible assurance about the absence of clandestine nuclear operations. In addition, the IAEA should be sure about the state's long-term political motivations. The biggest problem of the classical safeguards approach identified so far was the need for a continual verification of the DI including the declared mode of operations to cover the diversion scenarios "undeclared tunnelling" and "undeclared reprocessing below ground". If a state evaluation covering technical indicators as well as the overall political motivations arrived at the conclusion that clandestine reprocessing operations could be ruled out in the short and medium term, these two diversion scenarios, which involve a major effort in terms of time, materials, work and budget did not need to be covered by verification activities on-site, and verification needs could be drastically reduced.

Furthermore, the IAEA would have to take into account, whether in a country the media were controlled by the government or were independent. Another aspect was the country's legal system and its independence of the government. Further aspects were the existence of missiles and the features of the country's past nuclear programme. While there was agreement on the persisting importance of nuclear material accounting, the strength of such a system with social indicators would depend on the availability and credibility of sufficient open source information. The system would also depend on the IAEA's ability to put the results into relationship with the results gained from safeguards inspections.

These views were not shared by some experts who referred to the "Conceptual Framework for Integrated Safeguards" approved by the IAEA Board of Governors and to the geological repository safeguards approach recommended by the 1997 IAEA Advisory Group Meeting. The generic safeguards approach states the need to detect undeclared repackaging and undeclared reprocessing, to verify the inventory and its changes above ground, to maintain the continuity-of-knowledge on the repository boundary, to verify all transfers from above to below ground, and to detect undeclared retrieval of nuclear material from below ground. With a continuous monitoring system in place and within the criteria established by the Conceptual Framework, reductions in safeguards effort could be obtained through reduction in the frequency of data evaluations and the frequency or number of random inspections.

Some experts questioned the possibility to render technical systems failsafe even if applying the redundancy principle and improving the systems' reliability; it might lead to a loss of continuity-of-knowledge. Other experts stated that this issue was recognized in the development of the generic safeguards approach and that a well-designed site-specific safeguards approach using rugged, reliable, and redundant technical systems ensured continuity-of-knowledge. In such a safeguards approach technical systems complemented inspector activities and information analysis. Some experts insisted, however, that the state level approach with the criteria of plausibility and credibility would be preferable to continuous monitoring. No agreement was reached on resolving this issue.

It should, however, be recognised that the IAEA's safeguards culture is currently undergoing a change. Hitherto, IAEA safeguards implementation and evaluation were strictly based on safeguards criteria. In the future, the approach will be more flexible and country-specific, i.e., based on technical, institutional, and political characteristics of the state in question. It will provide a basis for assessing both anticipated inspection effort and subsequent evaluation.

7. Summary and Conclusions

At the meeting of experts held in Germany in June 2004, the following conclusions were drawn. Each repository will be individual in terms of the site-specific geological formation, the technical disposal

concept, and the state-specific implementation strategy. The underground structures of repositories will be rather complex, and safeguards inspectors will have to be provided means to keep their orientation.

A majority of experts endorsed the proposal to define the repository site by the security-fenced area above ground and the walls of the excavations below ground including all backfilled emplacement drifts and tunnels. However, no principal agreement was reached on the resolution of the repository site issue.

For design information verification (DIV) satellite imagery, ground penetrating radar (GPR), and laser range scanning might be feasible methods which have to be studied on a case-by-case basis.

For the detection of undeclared activities radionuclide analysis, passive seismic and acoustic monitoring, and satellite imagery might be feasible methods which have to be studied on a case-by-case basis.

Independent of the most advanced repository projects, the following programme activities should be carried out by the group of experts during the next years:

- Evaluation of the impact of Additional Protocol measures with regard to enhancing safeguards approaches and reducing verification needs;

- proposal of guidelines concerning the scope and evaluation of
  i. baseline information about the geological formation, and
  ii. design information related to the underground facility;

- proposal of feasibility studies of techniques for
  i. design information verification, and
  ii. detection of undeclared activities;

- help in providing expertise to the IAEA.

8. References

[1] Richter, B.; Report on the 7th Meeting of the Experts Group on Safeguards for Final Disposal of Spent Fuel in Geological Repositories; German Support Programme to the IAEA, report no. JOPAG/06.04-PRG-347; Forschungszentrum Jülich GmbH, D-52425 Jülich, Germany, June 2004.

[2] Okko, O., Rautjärvi, J.; Safeguards for the Geological Repository at Olkiluoto in the Pre-operational Phase; Finnish Support Programme to the IAEA, report no. STUK-YTO-TR208; STUK Radiation and Nuclear Safety Authority, FIN-00881 Helsinki, Finland, May 2004.

[3] IAEA Advisory Group Meeting on Safeguards Related to the Final Disposal of Nuclear Material in Waste and Spent Fuel (AGM-660), IAEA STR-243, Vienna, 1988.

[4] Model Protocol Additional to the Agreement(s) between State(s) and the International Atomic Energy Agency for the Application of Safeguards; INFCIRC/540 (corrected); A-1400 Vienna, Austria, September 1997/December 1998.

[5] Study on the Retrievability of Spent Fuel from a Closed Geological Repository in Rock Salt; German Support Programme to the IAEA, report no. JOPAG/12.96-PRG-273; by H.J. Engelmann et al., Forschungszentrum Jülich GmbH, D-52425 Jülich, Germany, December 1996.

Safeguards for the Finnish geological repository for spent fuel

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Abstract:

Owing to the early decision to focus on geological disposal of nuclear fuel, geological site investigation are carried out in Finland since early 1980'ies by the nuclear power companies to locate safe repository site. The final disposal of highly radioactive spent nuclear fuel in the geological repository at Olkiluoto was accepted be the local municipality, State Council and finally endorsed by the Parliament of Finland in 2001. The implementer's present plan is to construct an underground rock characterisation facility called ONKALO, consisting of a ramp, tunnel and shaft. The objective is to locate and design rock volumes that fulfil public safety requirements of the disposal in Olkiluoto. This pre-nuclear installation is not yet subjected directly to the regulatory control provisions of the Finnish Nuclear Energy Act nor of the Euratom Regulation before license is applied for construction of the nuclear disposal facility which is supposed to consist of the encapsulation plant and the underground repository. However it is expected that the first underground access tunnel will be part of final disposal facility and nuclear material will be moved through ONKALO tunnels to be emplaced in the bedrock. As a consequence, the National Safeguards System is set up in order to fulfil State's obligations based on the Non-Proliferation Treaty of Nuclear Weapons already at this early phase.

Keywords:final disposal of spent fuel, geological repository, continuity of knowledge

1. Introduction

The decision to construct an underground repository for spent nuclear fuel generated the development new types of geophysical activities in Finland in the 1980's. Site characterisation based on borehole logging of slim borehole brought in the need to develop new instruments and interpretation techniques to quantify the intactness and rigidity of the bedrock for the long-term safety assessment of the repository environment.

Based on information from geological site investigations and according to a democratic decision made in 2001, the final spent nuclear fuel repository shall be located at the Olkiluoto site in Eurajoki, western Finland. The next phase of site investigations includes the construction of underground premises, for rock characterisation purposes. The excavation of these galleries began in 2004. Later on, these premises are scheduled to form a part of the final repository if rock volumes are considered to fulfil safety requirements.

The final disposal of the nuclear material shall introduce new safeguards approaches which have not been applied previously in the International Atomic Energy Agency's (IAEA's) safeguards for spent fuel. The encapsulation plant to be built at the site will be the final opportunity for verification of spent fuel assemblies prior to their transfer to the geological repository. Moreover, additional safety and safeguards measures are being considered for the underground repository [1]. Integrated safeguards verification systems will be based on the verification of the excavated rock volumes as declared, but most probably geophysical monitoring will be applied to decline unannounced activities related to possible reprocessing or diversion schemes of nuclear materials at the repository site.

In order to support the development and implementation of the regulatory control of the final disposal programme, the Finnish Radiation and Nuclear Safety Authority established a national expert group [2]. The application of geophysical exploration methods as internationally accepted verification methods evokes the need to have accepted standards and proven technologies to fulfil the regulatory requirements. Therefore, the present DIV-technologies applied for safeguarding the repository are based on the standard surveying used as a part of the rock characterisation process.

2. Rock characterisation process and its verification

The decision to investigate Finnish bedrock for its suitability for the final disposal of spent nuclear fuel in 1983 started a large innovative phase of developing and applying a wide range of geophysical methods [3]. Innovative methodologies were developed to apply exploration geophysical methods in engineering, i.e., rock characterization in a quantifying manner. In these site investigations for nuclear waste disposal at several locations in Finland, the most hazardous parts of crystalline rock, i.e., water-bearing fractures were characterised and evaluated by several means. The first phase of these geophysical investigations at Olkiluoto was summarised by Heikkinen et al. [4]. Airborne, surface, and borehole geophysical methods were evaluated for their site-specific benefits and limitations.

The site-investigations continued at several locations until the Decision-in-Principle was made by the Finnish society and endorsed by the Parliament to accept the solidity of the crystalline rock at Olkiluoto to fulfil the confidence requirement to be the safe location for the burial of highly radioactive material. Before the underground excavations started, the hydrogeological, hydrogeochemical, rock mechanical, tectonic and seismic conditions were documented in a baseline report to serve as a reference point [5]. These characteristic were considered relevant to the long-term safety of the repository to be constructed. Since some changes or fluctuations in these characteristics are expected due to underground construction work, a monitoring program [6] is planned and scheduled to fit in the activity plan.

The final disposal of spent nuclear in geological formations is expected to serve the overall good of Finnish society. In this context, the national safeguards approach [7] of applying long-term safety data for safeguards in a cost-effective and non-intrusive manner according to IAEA guidelines [8] was developed and launched in 2003. The main focus has been on the generation of credible regulations for documenting construction and adjoining geoscientific monitoring records that have to survive over the 100-year disposal project. The timely documentation of the planned [9] and, in particular, the measured excavated, and later back-filled rock volumes is supposed to generate the Design Information declarations to be verified as safeguards measures during the operational time of the repository.

The monitoring program of Posiva carried out by several independent sub-contractors should bring in all the necessarily data to make safety-related observations if unexpected phenomena are located in the geological formation near the repository. Therefore, the possible observations in reduced safety are proposed be used as national safeguards alarms. Although, the interpretations of independent remote sensing or geophysical sounding is always a result of subjective survey planning, performance and reporting; and thus subject for argumentation, the reliability level of the methods applied should also satisfy the threshold level in reliability needed for conclusive safeguards verification measures. Owing to their nature and the indirect interpretation procedures, the indirect methods can be treated as scientific reports, but not as declarations to be verified. However, the present national requirement is to analyse the micro and macro seismic records collected for the long-term safety analysis also for their safeguards-relevance. The presence of the remote sensing instrument is supposed to have their deterrent influence.

The safety and safeguards issues are also under international reviews and inspections. Therefore, the safe disposal of nuclear materials is an issue of homeland security where the understanding of the geological site and the societal context play an important role. In order to facilitate independent international review and control, the IAEA has expressed a need to develop new site-specifically applicable geophysical methods for the IAEA's verification toolbox [10]. Nevertheless, the shared knowledge and site understanding creates the required confidence and trust in safeguards and, with no impact to the safety envelope, if so desired. The understanding of the site-specific knowledge and the uncertainty in exploration techniques and rock modelling should be openly addressed by the

parties involved both in geosciences and safeguards in order to facilitate the use of the new methodologies in safeguards.

3. Legal framework - a disclaimer

The documentation provided by Posiva [e.g. 5 and 6] is written by persons having their commitment in securing the long-term safety of persons, environment and property above the geological repository. Therefore, the terminology in references (and partly also in this text) has to be understood in the correct way: “the Underground Rock Characterisation Facility”, named by Posiva is not a facility subject to traditional safeguards since no nuclear materials are present at this “non-nuclear installation” as understood in safeguards [11]. Similarly, the repository site or the geological site investigations do not refer to a site delimited under the Additional Protocol (INFCIRC 540). The Basic Technical Characteristics (BTC) and the consequent Design Information (DI) are to be provided by the European Commission to the IAEA at the time of the nuclear commissioning of the disposal processes and installations approximately by the year 2020. Owing to these realities, the ONKALO is declared by Finland as a general plan relevant to the development of nuclear fuel cycle 2a(x) under the Additional Protocol.

Referring to the recommendations generated in the International Atomic Energy Agency’s Programme for Development of Safeguards for Final Disposal of Spent Fuel in Geological Repositories [1], STUK established (in 2002) and operates the National Competence Network [2 and 7], including the implementing company and other relevant institutions, in order to fulfil State’s obligations based on the Non-Proliferation Treaty of Nuclear Weapons already at this early phase of repository development. The National System is to generate the safeguards-relevant data base to be provided as a part of BTC and DI at the time of licensing and commissioning the nuclear facility. Moreover, co-operative negotiations are continuing between the State representative (STUK) and the IAEA to define the actual data and information requirements and to establish inspection framework during the present pre-nuclear phase.

4. Conclusions

The final disposal of the nuclear material shall introduce new safeguards approaches which have not been applied previously in the IAEA’s safeguards for spent fuel. The present stage of underground construction can be verified through human access and by use of geodetic underground surveying techniques. This requires continuous presence at the installation. The cost-effective and non-intrusive national safeguards system is based on the timely validation and verification of the underground geological site investigation process. The excavation has been underway since autumn 2004 and the parties are encouraged to collaborate in the implementation of the required measures.

In the future repository, there will be no direct methods to verify the disposed nuclear fuel. Therefore, the safeguards approach will be based on the measures that will effectively utilise information and data from safety-related confidence-building processes. New remote sensing and geophysical techniques may assist in maintaining a coherent picture about the socio-technical complex subject to safeguards, providing also for desired deterrence.

5. References

[1] International Atomic Energy Agency, 1998. *Safeguards for the Final Disposal of Spent Fuel in Geological Repositories*. IAEA, STR-312, 5 Volumes.

[2] Okko, O. (ed.) 2003. *Safeguards for Final Disposal of Spent Nuclear Fuel. Methods and Technologies for the Olkiluoto Site.* Radiation and Nuclear Safety Authority, STUK-YTO-TR 199, 59 p.

[3] Poikonen, A. 1983. *Suitability of certain borehole geophysical methods for structural and hydro-geological studies of Finnish bedrock in connection with disposal of nuclear waste*. Nuclear Waste Commission of Finnish Power Companies, Report YJT-83-06. 59 p.

[4] Heikkinen, E. (ed.) 1992. *Geophysical investigations in the Olkiluoto area, Finland. Summary Report*. Nuclear Waste Commission of Finnish Power Companies, Report YJT-92-34, 110 p.

[5] Posiva, 2003. *Baseline conditions at Olkiluoto.* Posiva Report 2003-02, 220 p.

[6] Posiva, 2003. ONKALO*, Underground Characterisation and Research Programme*. Posiva Report 2003-03, 142 p.

[7] Okko, O. & Rautjärvi, J. 2004. *Safeguards for the Geological Repository at Olkiluoto in the Pre-Operational Phase*. Radiation and Nuclear Safety Authority, STUK-YTO-TR 208, 30 p.

[8] International Atomic Energy Agency, 2001. *Monitoring of geological repositories for high level radioactive waste*. IAEA-TECDOC-1208 25 p.

[9] Posiva, 2003c. ONKALO*, Underground Rock Characterisation Facility - Main Drawings Stage*. Posiva Report 2003-26, 71 p.

[10] International Atomic Energy Agency, 2000. *Report of the experts' meeting on the use of geophysical techniques for safeguarding geological repositories*. IAEA, STR-324, 29 p.

[11] International Atomic Energy Agency, 2002. *IAEA Safeguards Glossary*. 2001 Edition, 218 p.

Session 14

Training and knowledge

Addressing the Demographic Time Bomb

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Abstract:

Nuclear agencies worldwide currently face a "demographic time bomb" – the reality that, as the "post-war" generation reaches retirement age, an unusually large percentage of the experienced safeguards workforce will retire within the next several years. This looming crisis poses the challenge of providing a large cadre of new inspectors with sufficient training and on-the-job support to maintain safeguards activities at the expected high level.

Partially as the result of this demographic crisis, a second challenge has emerged. Traditional training approaches alone are no longer sufficient to achieve the required transfer of knowledge. For instance, the rigorous work and travel schedules of experienced inspectors often make their participation in traditional classroom-based training difficult or impossible. Similarly, the traditional use of "mentoring" – the pairing of skilled and less skilled persons with the goal of helping the lesser skilled person develop specific competencies – has suffered from the difficulty of getting busy, highly mobile individuals together at the same time and place.

This paper reviews recent initiatives by the Canadian Safeguards Support Program (CSSP) to address the knowledge transfer challenges arising from the demographic time bomb, and examines some emerging approaches that show particular promise. More specifically, it looks at the use of low-tech, low-cost information capture techniques to preserve and transfer corporate knowledge from experienced professionals to their replacements. It also explores the emerging capabilities of E-mentoring – which describes mentoring that takes place over a distance, usually by means of the telephone and Internet. E-mentoring appears to work particularly well when the less experienced partner needs specific instructions or explanations, as is often the case in the safeguards environment.

Drawing on recent experiences in capturing corporate memory from experienced IAEA inspectors, and the E-mentoring experience of two Canadian federal government departments – Health Canada and Natural Resources Canada – this paper describes the basic building blocks and strategies of successful knowledge transfer and E-mentoring programs, defines the value proposition for doing so, highlights lessons learned, and explores how these innovations could be used to change safeguards learning and work in the future.

Keywords: corporate memory; e-mentoring; information capture; knowledge transfer; mentoring; nuclear safeguards; training

1. The Demographic Time Bomb

Nuclear agencies worldwide currently face a "demographic time bomb" – the reality that, as the post World War II generation reaches retirement age, an unusually large percentage of the experienced safeguards workforce will retire within the next several years.

For instance, at the International Atomic Energy Agency (IAEA), a significant number of its most senior safeguards personnel are expected to retire within the next five years. Attrition rates at the Unit Head level and below are expected to be even higher. At the IAEA the issue is compounded by a fixed retirement age and by a staff rotation policy that generally limits the duration of employment within the organization.

In the Canadian nuclear industry, key organizations expect 35% or more of their experienced people to leave over the next 8 years, with the peek occurring in the 2006-2007 time period. That's the *best* case scenario. The concern is that some eligible retirees will delay their retirement by a few years causing departures to bunch up, or that the emergence of retirement incentives could cause early departures in greater-than-expected numbers. Worries about the timing of retirements arises, in part, from the very long recruitment period for replacement personnel. In certain situations, it takes up to 18 months to find and hire a suitable employee. If hiring cannot be reliably synchronized with expected retirements, capabilities may be temporarily eroded.

The demographic crunch comes at a time when the increasing complexity of systems and methodologies employed by the nuclear industry requires that inspectors have ever-greater levels of knowledge and skills. Together, these two trends pose significant challenges.

First, there is the considerable challenge of finding qualified replacements for retiring nuclear industry professionals. This is a problem both because the "Generation X" workforce is smaller than its predecessor – leading to increased competition for the available labour – and because science and engineering schools around the world, with few exceptions, are no longer preparing their graduates specifically for careers in the nuclear industry.

Second, assuming that a sufficient number of replacement workers can be hired, safeguards agencies such as the IAEA must face the challenge of providing a large cadre of new, inexperienced inspectors with sufficient training and on-the-job support to maintain safeguards activities at the high level of competence expected. Solutions to this challenge will be examined in greater detail in this paper.

2. A Crisis in Knowledge Transfer

The challenge of meeting the training and on-the-job support needs of inexperienced employees runs headlong into yet another emerging reality. Traditional training approaches alone are no longer sufficient to achieve the required transfer of knowledge.

Traditionally, instruction has taken place at the commencement of employment and periodically thereafter in training facilities that are physically removed from job sites. In addition to classroom instruction, safeguards agencies have endeavoured to provide employees with opportunities for on-the-job learning. This has consisted mainly of ad hoc mentoring and coaching, access to procedural manuals, occasional telephone support from subject matter experts, and, in a growing number of instances, to on-line access to headquarters reference materials by means of secure networks.

In normal circumstances, these approaches might be sufficient. However, throughout the nuclear industry – and, indeed, in all industries in developed countries – these traditional methods are experiencing extraordinary stresses. New procedures, new safeguards devices and a heightened awareness of the crucial importance of compliance regimes demand more knowledge, training and rigour in carrying out inspection routines at a time when subject matter experts and experienced trainers are in short supply, training budgets are constrained, and workloads are increasing.

For instance, the rigorous work and travel schedules of experienced inspectors often make their participation in traditional classroom-based training difficult or impossible. Similarly, the traditional use of "mentoring" and "coaching" – the pairing of skilled and less skilled persons with the goal of helping the lesser skilled person develop specific competencies – has suffered from the difficulty of getting busy, highly mobile individuals together at the same time and place.

Under these circumstances, it is understandable that the search is on to find better ways to make training, information and on-the-job support of all kinds available to nuclear industry professionals in new, more convenient, more effective and less costly ways.

3. Towards a Knowledge Transfer Strategy

Organizations of all kinds are accustomed to developing written strategies for the achievement of their goals. For instance, many organizations develop long-range business development strategies, communications strategies, and training strategies. In recent years, the realization has gradually emerged that the same kinds of processes and frameworks can be used to develop a *knowledge management* or *knowledge transfer* strategy – a plan for the organized identification, evaluation, capture, preservation, organization, enhancement and transfer of knowledge assets within an organization. The objective of such a strategy is to ensure that knowledge flows from wherever it may exist throughout an organization, by whatever combination of means makes the most sense, to the people who need it to perform their tasks at a high level of skill. A closely related objective is to leverage knowledge assets to encourage innovation and to deal with emerging problems quickly and effectively.

A growing body of experience in implementing knowledge management or knowledge transfer strategies has helped to identify several key "building blocks" of an effective framework. These include: management support; a clear understanding of the organization's knowledge requirements; identification of existing knowledge assets and gaps, a knowledge transfer architecture; a positive culture; and support personnel. If any of these "blocks" is missing, the chances of sustaining effective knowledge transfer in the long run are reduced substantially, as each element lends strength to the others.

4. Addressing "Time-Bomb" Challenges

In recent years, the Canadian Safeguards Support Program (CSSP), which operates within the Canadian Nuclear Safety Commission (CNSC), has taken on the task of identifying the elements of a multi-faceted knowledge transfer framework, and of experimenting with a variety of specific approaches and perspectives. It is carrying out this work as an initiative in support of the International Atomic Energy Agency (IAEA).

4.1. Technology-Assisted Instruction

The CSSP's first initiatives in this area began in the late 1990s (see Ref. [2]). Initially, the focus was on using technology-assisted instructional techniques to provide just-in-time, on-the-job training for field inspectors. For instance, CD-ROM-based instructional programs were developed to familiarize field inspectors with the features and layout of CANDU power plants, and to teach them how to use an advanced-technology fuel monitor now installed at CANDU sites. These programs, which inspectors can take with them to review on their laptop computers as required have proven to be highly successful in providing high quality instruction, reducing training times, and cutting associated training costs.

In the same period, the CSSP began testing impromptu video and audio recordings as a means of capturing presentations by subject matter experts who might not be able to attend conventional classroom sessions (see Ref. [3]). The recordings were digitized and made available on CDs and, more recently, on the Web. Video animation was also used to illustrate processes that cannot be observed directly, such as refueling procedures or the operation of Core Discharge Monitors. All of these techniques proved to be useful and have remained in use to the present day.

4.2. Towards a Management Framework

To help give context to these individual techniques, the CSSP began in 2002 to look more broadly at management frameworks for knowledge transfer by surveying knowledge management programs and best practices in Canadian federal departments and agencies. The search identified a wide range of knowledge mapping, preservation, and transfer activities, including skills inventories, corporate information portals, e-learning initiatives, and the gathering of metrics to help track the impacts of knowledge management activities on organizational performance.

Since that time, the need for nuclear knowledge management systems has been elevated to a central position in the thinking of key nuclear organizations around the world. For instance, in an invitation recently issued by the IAEA to attend the Technical Meeting and Workshop on Managing Nuclear Knowledge that will take place in Trieste, Italy, in late August of this year, the Agency describes knowledge management as "representing the primary opportunity for achieving substantial savings, significant improvements in human performance, and competitive advantage."

Some of you may be planning to attend the First International Conference on Nuclear Knowledge Management, sponsored by the World Council of Nuclear Workers, which will take place in Santos, Brazil, in late August and early September. In these conferences and others, the purpose is to share knowledge management information and strategies as a way to bring a new level of sophistication to this crucial activity.

4.3. Emerging Knowledge Transfer Techniques

Previous papers presented by the CSSP have dealt with the "big picture" subject of knowledge management – and it remains a key interest (see Ref. [5]). However, the main purpose of this paper is to draw attention to some emerging knowledge transfer techniques that fit well within the bigger picture.

In this regard, solutions in which the CSSP has recently taken an interest include:

E-doing: the integration of learning and on-the-job support by providing a single on-line portal through which employees have access to just-in-time instructional modules, a wide range of easily searchable reference materials, as-needed advice and coaching from subject matter experts, diagnostic and decision-making tools, document templates, and utilities such as discussion forums and chat rooms to communicate and share information with colleagues (see Ref. [4]);

Automated Information Capture: the use of sophisticated software tools to assist in the capture, indexing, retrieval and planned retirement of corporate information resources;

Communities of Practice: the use of on-line discussion techniques to support individuals who practice the same craft and face similar challenges arising out of the craft (i.e. facility inspection). The focus is on *practice* -- on trying to improve how the members perform the work they do; and

Virtual Office Solutions: using software to create a common work environment in which files can be shared, agendas coordinated, and discussion threads preserved. One of the most widely used solutions of this type is the wiki, a collaborative Web site comprised of the perpetual collective work of many authors (see Ref. [8]). A "wiki" – which means "quick" in Hawaiian – allows anyone to edit, delete or modify content posted by any of the contributors. Another solution is an on-line workspace tool such as Groove or NetMeeting.

5. An Emerging Technique: E-Mentoring

An approach called *e-mentoring* has also been identified as a promising technique for addressing the knowledge transfer challenge. Given the increasing difficulties of getting busy, highly mobile mentors, coaches and associates together at the same time and place, the development of an effective virtual alternative is a welcome development.

E-mentoring is about one person (a mentor) sharing his or her knowledge to help another (the protégé), and, in so doing, enriching the protégé, mentor and their organization. The approach enables mentors and protégés to communicate with each other via telephone, e-mail, personal discussion forums, or a combination of these techniques. E-mentoring is especially useful in making a good match between an experienced mentor located at one site, and a protégé located at a different site often hundreds or thousands of miles away. A second distinguishing feature of e-mentoring is the use of on-line software to streamline the recruitment, matching, training, and support of mentorship partners. This greatly reduces the administrative burden of running a mentorship program.

In the Canadian federal government experience, where several programs of this kind have been implemented, the consensus view is that e-mentoring should not substitute entirely for face-to-face mentoring. Typically, an e-mentoring relationship is sought on-line but, when established, is followed as soon as possible by an initial meeting in person between mentor and protégé. This face-to-face relationship is renewed periodically as the partners' schedules permit.

Two current examples will help to demonstrate the operation and value of the e-mentoring approach.

5.1. E-Mentoring at Natural Resources Canada

At Natural Resources Canada (NRCan), e-mentoring began to replace traditional face-to-face mentoring and coaching as a tool for direct person-to-person knowledge transfer in the late 1990s.

NRCan is the federal government department specializing in the sustainable development and use of natural resources, energy, minerals and metals, forests and earth sciences. The department brings expertise in science and technology, policy and programs to the management of these resources on behalf of Canadians. Headquartered in Ottawa, the department has Regional Offices and laboratories across Canada, which creates significant communication and coordination challenges.

For many years, the department has used conventional approaches to mentoring and coaching as a means of introducing new employees to their assigned tasks and of increasing their level of knowledge and skills over time. Until the 1990s, the majority of scientists and other professionals were located centrally in the National Capital Region. At that time, the situation began to change as a significant proportion of the department's employees were dispersed to Regional Offices and laboratories. This change hampered traditional approaches to mentoring because potential mentors and protégés were often in widely separated locations.

In the late 1990s, NRCan's Human Resources Branch began exploring Internet-based approaches as an alternative to conventional mentoring. More specifically, they looked to on-line technology both as a means of matching mentorship participants, and of sustaining their relationship over time.

In order to reduce the workload required of human resources managers to identify and match potential mentors and protégés, a custom-designed match-making program was developed to make this function largely self-supporting. The existence of the e-mentoring match-making service is promoted throughout the department using posters, brochures, periodic broadcast emails, mention during new employee orientation training, and so on. However, it is left largely up to individual employees to register as participants.

Those who are interested log-on to an Intranet-based site, where each potential candidate creates a personal profile using a set of predetermined criteria that includes topics for discussion and category of work. An automated matching process then provides participants with a list of potential NRCan mentors or protégés, based on the criteria entered, regardless of where they are located in Canada.

Candidates have the option of self-identifying as one of four employment equity groups and/or indicating a desire to be matched with a person who has self identified as an employment equity group. The groups include women, members of visible minorities, persons with disabilities, and Aboriginal peoples.

Once matched, participants in the mentoring relationship can then look forward to support from the organization in terms of books, periodicals, and a workshop designed to hone mentoring skills. The half-day workshop, which is conducted at local sites across the country as required, provides participants with some of the essential information and skills needed for them to have effective and productive interactions with their mentor or protégé. In support of the e-mentoring program, this session assists participants in exploring the power of mentoring and the positive consequences for both mentors and protégés.

In its six years of existence, the NRCan e-mentoring approach has shown promise as a method of ensuring that the appropriate knowledge, skills, and attitude are present for the mentoring relationship to flourish. At present, the program is going through a redesign phase to update the supporting software and, in the face of declining numbers of participants during the past year, to rethink the department's approach to recruiting mentors and protégés. However, the overall assessment of the e-mentoring approach is that it has proven its worth as a tool for nourishing careers and transferring essential knowledge from one generation of employees to another.

5.2. E-Mentoring at Health Canada

At Health Canada, the Mentoring Program is administered by eight regional Career Centres in major cities across the country. At the national level, the program is championed by a Director-General, Human Resources, and who promotes the program to her senior management colleagues and lends credibility to it among employees.

Designed to supplement career management services offered by the Career Centre, the Mentoring Program aims to provide a bank of mentors who may be called on to advise employees. Formalizing the mentoring activity ensures that such assistance is available to any employee who believes that he or she could benefit from this kind of assistance.

All experienced managers who wish to participate in the Mentoring Program, after completing a registration form, are interviewed to discuss their particular interests, strengths and availability. A training program on mentoring is offered to all mentors and potential mentors. Mentors have been more difficult to recruit than protégés, so there is a perceived need for extra effort in this regard. To meet a current imbalance, some mentors have more than one protégé.

Employees who wish to access the Mentoring Program do so by completing a registration form on which they are asked to identify clearly their expectations from a mentoring relationship. They are also met by a Career Centre representative to discuss their expectations. Based on the information provided by the employee, the Career Centre identifies a potential mentor, contacts the mentor and then with his/her permission provides the employee with this information. The protégé is then asked to contact the mentor directly to arrange a first meeting.

Periodically, the Career Centre monitors the success of the mentoring activity by means of a feedback form filled out jointly by the mentor and protégé, periodic lunch-and-learn sessions in which protégés are invited to share their experiences with others, and informal discussions with individual participants. However, once a mentoring partnership has been established, responsibility for its on-going operation and health rests largely with the partners.

As with Natural Resources Canada, the Health Canada program pre-dates the introduction of technology-assisted mentoring methods. Although face-to-face meetings between mentors, protégés and Career Centre coordinators remain an important feature of Health Canada's program, many interactions now take place by telephone, email, or other distance means agreed on by mentoring partners due largely to the physical separation of many mentoring pairs. Health Canada, like Natural Resources before it, is looking seriously at creating an e-mentoring framework, perhaps based on the Web-based *Open Mentoring*™ software solution from Triple Creek Associates (www.3creek.com).

5.3. Incentives for Mentors

A question that often comes up is: what's in it for the mentor? For protégés, the advantages seem obvious. A mentoring relationship is an opportunity to obtain sound advice, guidance and encouragement, and to access organizational knowledge from an experienced practitioner. But for the mentor, isn't it just another burden for an already busy person – especially as these programs are rarely accompanied by special rewards such as extra holidays or financial rewards?

In fact, there are a number of incentives -- personal and institutional -- for becoming a mentor. Putting aside for the moment the altruistic goals of helping the organization to succeed and taking pleasure in seeing youth blossom, a sound personal reason for taking on mentoring responsibilities might be the need to have competent people to whom work can be delegated. By taking the time to help a junior colleague develop his or her skills, the mentor can often reduce workload in one area in order to concentrate efforts elsewhere. Mentoring can also be about creating a circle of trusted coworkers -- a support network that can be important to survival and peace of mind in the corporate world. An example of an institutional reason for taking on a mentorship is a performance accord that requires an executive to groom possible successors. Mentorship is one way of fulfilling that commitment.

Some mentorship systems make use of simple rewards such as public recognition or hosted luncheons for mentorship partners attended by senior executives. These and other acts of recognition are often very well received.

For the organization as a whole, the benefit is getting more knowledgeable employees with broader perspectives. Mentoring also promotes enhanced communication and the sharing of values, which contributes to a more motivated and effective workplace.

6. The Value Proposition for E-Solutions

Organizations that have adopted forward-looking solutions of all the types just described – e-learning, e-doing portals (see Ref. [4]), communities of practice, e-mentoring, and more – are generally satisfied with the results for some or all of the following reasons:

- Most of these approaches link learning and information resources directly with work. This improves employee performance by providing assistance when a task is undertaken.
- The approaches give employees increased control over their own learning. In today’s workplace, employees are expected to manage and develop their skills and employability.
- E-solutions provide flexibility. Employees don’t have to fit themselves into pre-scheduled classes that may cover more than they need to know at present. Instead, they can receive modules of information and learning that fit their current needs, or benefit from an on-going relationship with a mentor who is a continuing source of information, advice and motivation.
- E-solutions encourage information sharing, collaboration and interaction. This is an important benefit for team building and creating a sense of shared purpose.
- The organization can achieve significant savings. It is cost-effective to use existing information and communications technologies to deliver training and support. In addition, significant savings stem from reduced travel expenses, less employee time off work, and lower expenditures on other forms of training and communication (many organizations that have implemented e-learning or Web-portal on-the-job support solutions report savings of 50-80% over previous expenditures).
- E-solutions for knowledge transfer, if properly implemented, result in improved learning outcomes and productivity, and reduced safety and compliance accidents and infringements.

It is particularly significant that many E-solutions are delivered by means of Internet or Intranet technologies. The main benefits that flow from the use of these media are access and flexibility: employee control over transactions; the ability to reach more employees in different locations; and just-in-time access to learning.

7. Conclusion

In the quest for new ways to meet the knowledge transfer challenges created by the demographic time-bomb, it is easy to lose sight of the need for balance between traditional, time-tested, face-to-face methods of conveying knowledge, and newly emerging approaches offered by computer- and Internet-based approaches. The attitude within the CNSC's Canadian Safeguards Support Program and in many other organizations is that the best strategy is to implement the best of both worlds, keeping the face-to-face activities that have proven to be so good at building relationships, teamwork, perspective, hands-on competencies, and loyalty – and complementing them with technology-assisted methods that span distances, provide instant refreshers, and facilitate searches for the right data, knowledge, and support tools at the right time.

The potential benefits identified in this paper support the case now being made by nuclear safeguards agencies in documents such as the IAEA's recent call for participation in the Trieste Meeting and Workshop on Managing Nuclear Knowledge (see Section 4.2). The demonstrated benefits support both adopting a more systematic approach to knowledge management, and making effective use of the wide selection of technology-assisted methods available. There is no single "magic bullet". It is important to see each technique – such as the e-mentoring technique highlighted in this paper – within the larger context of a multi-faceted knowledge management and transfer system.

In its work to assist the IAEA Department of Safeguards, the Canadian Safeguards Support Program will continue to identify and test ways to address the challenges posed by the demographic time-bomb.

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References

[1] Carroll, C.; Killeen, T.; Stein, M.; *The Impact of the High Inspector Turnover Rate on Safeguards Training*; Sonalysts, Inc., Paper presented at the INMM 42nd Annual Meeting, Indian Wells, CA, USA, 15-19 July 2001.

[2] Truong, Q.S. (Bob); Keeffe, R.; Ellacott, T.; Desson, K.; Herber, N.; *Inspector Training for VIFM Equipment: An Integrated Approach,* IAEA Sumposium on International Safeguards – Verification and Nuclear Material Security (Poster Session), Vienna, Oct. 29 – Nov. 2, 2001.

[3] Truong, Q.S. (Bob); Keeffe, R.; Desson, K.; *New Direction in Safeguards Training: Preserving Corporate Memory and Harnessing the Potential of Distributed Learning*, paper presented at INMM 43rd Annual Meeting , Orlando, FL, USA, 24-27 June 2002.

[4] Truong, Q.S. (Bob); Keeffe, R.; Desson, K.; *Integrated Learning and Doing – Single Portal Access to Instruction, Decision Making Tools and Information Sharing*, paper presented at INMM 44th Annual Meeting, Phoenix, AZ, USA, 13-17 July 2003.

[5] Truong, Q.S. (Bob); Keeffe, R.; Desson, K.; *Creating the Conditions for Effective Safeguards Knowledge Transfer – Building Blocks of an Effective Knowledge Transfer Framework,* paper presented at INMM 45th Annual Meeting, Orlando, FL, USA, 18-22 July 2004.

[6] Truong, Q.S. (Bob); Keeffe, R.; Herber, N.; Barroso, H.; *Just-In-Time Knowledge Capture Techniques*, paper presented at INMM 45th Annual Meeting, Orlando, FL, USA, 18-22 July 2004.

[7] DeLong, David W.; *Lost Knowledge: Confronting the Threat of an Aging Workforce*, Oxford University Press, Oxford, 2004.

[8] Truong, Q.S. (Bob); Herber, N.; Barroso, H.; Liguori, C.; *Low-Cost Knowledge Management Techniques*, paper to be presented at INMM 46th Annual Meeting, Phoenix, FL, USA, 11-14 July 2005.

[9] Truong, Q.S. (Bob); Keeffe, R.; Desson, K.; *Getting the Balance Right: Integrating IT and Person-to-Person Knowledge Transfer Techniques*, paper to be presented at INMM 46th Annual Meeting, Phoenix, FL, USA, 11-14 July 2005.

From Training towards Education in Safeguards[1]

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Abstract:

After the establishment of EURATOM and the International Atomic Energy Agency nuclear inspectors had to be trained. The Western safeguards methodology was soon recognized as an international standard and has been disseminated to Eastern countries such as the Newly Independent States. Nowadays a large variety of safeguards training courses are offered at the JRC sites in Ispra and Karlsruhe, at the US National laboratories, at the Russian Methodological Training Center in Obninsk, beside other nuclear organizations for expert courses.
However, for University students the safeguards terminology remains undefined. Academic education in nuclear engineering or physics covers typically the following three disciplines:

- *physics of atoms with neutronics and with nuclear instrumentation*
- *reactor technology, with thermal-hydraulics of nuclear systems (inclusive nuclear safety aspects)*
- *radiochemistry and nuclear materials,, inclusive the nuclear fuel cycle*

Safeguards aspects fall normally out of the scope in order to avoid any political discussion. With the media reporting extensively on suspect countries not respecting the agreements under the Non-Proliferation Treaty, students have pertinent questions on safeguarding of nuclear material and technology. Moreover each case of illicit trafficking of nuclear or radioactive material highlighted by the press reveals a concern on nuclear security.
The BNEN (Belgian Nuclear Higher Education Network), a national representative of the European Nuclear Higher Education Network (ENEN), recognized the need to address safeguards aspects under the format of an advanced course. A first course of 3 ECTS points (15 hr theory) has been organized for this BNEN year 2004-2005 with about 20 students. Moreover the BNEN Steering Committee accepted a first diplome thesis in this area, which initiated in October 2004.
The ESARDA "Course Modules" initiative can and should play a key role in teaching the safeguards terminology and methodology to docents at Universities by providing them the relevant information in a concise manner. It is the aim to establish for academic staff online (freely available after registration) 5 generic course modules:

- *introduction with an overview on the legislation and the national control systems*
- *principles and logic of safeguards with an overview on the fuel cycle and on the facilities with nuclear material of civil/military origin*
- *nuclear material accountancy and control methods (with Statistics, Physical Inventory Verification, Near Real Time Accountancy)*
- *accountancy and verification measurements (with Non-Destructive Assay, Destructive Analyses, Containment/Surveillance, Mass/Volume techniques)*
- *Integrated Safeguards (with Additional Protocol, Open Source Information Technology and Satellite Monitoring)*

Keywords: Training, Education, safeguards terminology

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1. Introduction

Since the 1950's the first nuclear power plants have been constructed and the nuclear technology was developing exponentially. Together with the nuclear industry and in direct support to both the utilities and the national inspectors, research centers have been investigating safety aspects with deterministic and probabilistic studies, supported by various experimental tests. During the seventies design basic accident studies were carried out, and formed a part of the safety licensing report. In the eighties also severe accident analyses have been carried out and lessons learned from some accidents (such as the Three Mile Island accident).

In the mean time, based on the Non-Proliferation Treaty and reinforced in the nineties with the Additional Protocol safeguards aspects have been addressed mainly by research institutions in support to the states, owner of the nuclear material and to the international inspectors (EURATOM and IAEA for the European Member States). Nowadays with the threat of terrorist attacks (such as on 11 September 2001) strengthening of the nuclear security should be one of the priorities of the international community.

Understanding of the nuclear safety and safeguards aspects is very useful to address the nuclear security in an efficient way. Maintenance of nuclear knowledge, also by the younger generation is therefore required. Not only communication to the public in general and training of professionals but also the basic education should be provided. Nowadays at European universities the nuclear discipline is rather limited because of the lacking popularity. Typically the following three disciplines are only addressed:

- physics of atoms with neutronics and with nuclear instrumentation
- reactor technology, with thermal-hydraulics of nuclear systems (inclusive nuclear safety aspects)
- radiochemistry and nuclear materials,, inclusive the nuclear fuel cycle

Although nuclear safety is recognized to be an important academic discipline, safeguards has not such statute. The more politicians discuss safeguards topics, the more these topics are avoided in an engineering education. As universities are already less and less teaching nuclear courses, especially university courses on the nuclear fuel cycle become rare and the introduction of a new safeguards course improbable.

Beside the emphasis on education in nuclear safety also the importance of education in nuclear safeguards has to be clearly indicated. The broad span of influence and control, direction and managing of nuclear materials, regulation and implementation of safeguards systems exhibited by a large number of professionals, nuclear facility managers, technical specialists, inspectors, diplomats, jurists, politicians, researchers and teachers, etc. has to be recognized. Moreover, Dickman [1] recently stated that, given the rapidly evolving world climate since the end of the cold war, education and training is needed for safeguards leaders and experts with a well-developed understanding of the broader political dimensions of current non-proliferation challenges.

The younger generation receives, via the various media, a lot of information about the risks of proliferation. One of the objectives of safeguards education is to provide them with the necessary technical background information, validated by the professionals and practitioners, enabling to make their own judgment on the various information sources.

2. Current safeguards community needs to engage younger generation

Traditionally the nuclear safeguards and non-proliferation programme of the past second half of the 20th century focussed on treaties, security controls at export borders, nuclear material accountancy and control systems with regulations, instrumentation, data analysis, remote verification. Nuclear research centres steadily developed enhancements and modernisations in the various fields of Non-Destructive and Destructive Analyses, Solution Monitoring techniques and Containment Sealing and Surveillance Systems. Less emphasis has been placed on recognition of the human element as a primary component of the research infrastructure and the key to successful and sustainable implementation of safeguards programmes.

Until nowadays, the safeguards community was limited to a relatively small and homogeneous society of informed specialists. This was simplifying the decisions making process and was enhancing the efficiency by more direct communication paths between its members. Now more and more people from various horizons have to deal with enlarged

subjects and increased importance of the fields.

New European Union borders even highlight the interest in homogeneous teaching material aiming at providing "European" approach to the various publics concerned. This EU harmonized material would also facilitate in medium term, a European approach to the nuclear Non Proliferation.

The replacement of a first wave of researchers and more generally technical people facing retirement becomes problematic because of lacking young graduates, engineers, physicists or technicians in the nuclear field. Moreover the teaching community itself decreases in number.

3. Embedding in existing educational networks

The OECD recommends to develop "educational networks among universities, industry and research institutes" [2].

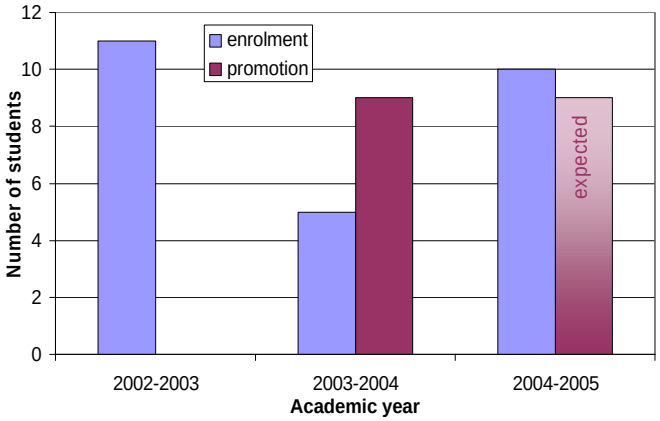

Fig. 1: BNEN students

The Commission supported and funded as EURATOM FP5 project the establishment of a roadmap for educational networks in the nuclear field. Such educational networks aim:

- guaranteeing nuclear knowledge and expertise through the preservation of higher nuclear engineering education
- optimally utilising the dwindling teaching capacity, scientific equipment and ageing research infrastructure through co-operation between universities and research centres

In 2001 a Belgian Nuclear Higher Education Network BNEN [3] was created with five Belgian universities (Vrije Universiteit Brussel, Katholieke Universiteit Leuven, Universiteit Gent, Université de Liège, Université Catholique de Louvain) and the Belgian Nuclear Research Centre SCKCEN as a joint effort to maintain and further develop a high quality programme of 60 ECTS (one full academic year [4]) for a "Master of Science degree in Nuclear Engineering". Fig. 1 gives and overview of the students for the master specialization in nuclear engineering that are enrolling respectively promoting. Under an FP6-action the outcome of this network will be evaluated by students, lecturers and a third neutral body in a final review report.

In parallel a similar network with a European dimension, the European Nuclear Higher Education ENEN [4], has been setup with a twenty-one universities and with the nuclear industry, regulators and research centers as stakeholders, mainly lead by INSTN/CEA. In the meantime this network has been growing towards the now so-called NEPTUNO network (Nuclear European Platform for Training and University Organisations) with 35 partners from Austria, Belgium, Bulgaria, Czech Republic, Finland, France, Germany, Greece, Hungary, Italy, Romania, Slovakia, Slovenia, Spain, Sweden, Switzerland, The Netherlands, United Kingdom. This European network is facing the second academic year and therefore no real statistical data can be shown. Up to now students enrolled and the first ENEN promotions are expected for the Summer 2005.

4. First experience with an academic course in Nuclear Safeguards

Upon request of the BNEN students a 3-days course on Nuclear Safeguards and Non Proliferation was organized at Ispra the first week of March 2005 as an Advanced Special Course with 3 ECTS in the BNEN programme. Eighteen students from various institutions (as shown in Fig. 2) attended the course. As safeguards topics: the legal basis, safeguards principles, accountancy auditing, destructive and non-destructive measurements, solution monitoring and containment surveillance have been addressed by seven lecturers from University Ghent, University Leuven, SCKCEN and from IPSC JRC Ispra and ITU JRC Karlsruhe during two full days. The third day was dedicated to the visit of laboratories and practical exercises. The schedule of the course

[4] ECTS: European Credit Transfer System: http://europa.eu.int/comm/education/socrates/ects.html

is given in Fig. 3. With the positive feedback, this course will be repeated yearly as part of the standard curriculum of the BNEN programme.

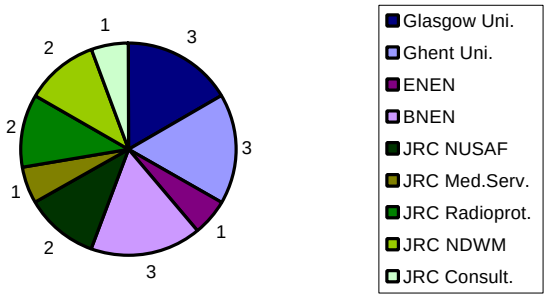

Fig. 2: Students of the first safeguards course

5. First dissemination of lecture notes

The lecture notes of the Ispra course, 1-3 March 2005, available to all participants, have been structured as follows:

- *introduction with an overview on the legislation and the national control systems: treaties & protocols, INFCIRC's*
- *principles and logic of safeguards with an overview on the fuel cycle and on enrichment, fuel fabrication and reprocessing facilities*
- *auditing of nuclear material accountancy (from statistical point of view, inclusive accuracy and frequency of Physical Inventory Verification, Near Real Time Accountancy)*
- *measurements for nuclear material accountancy and control (NMAC) with Solution Monitoring, Non-Destructive Assay (inclusive particle analyses), Destructive Analyses, Containment sealing and Surveillance Techniques)*
- *Integrated Safeguards (Open Source Information Technology and Satellite Monitoring as methods to verify compliance with the additional protocol) and illicit trafficking issues*

The organizers also archived the lecture notes on the MINERVA website[5] of the University of Ghent because students are nowadays used with on-line classroom and to download the course syllabus from the university internet with their userID and password. Exercises are becoming homework and the answers are commonly communicated by email. The interest in access via web and in computerised based training, such as expressed by the IAEA [5] is obvious.
Moreover to look up something, young people do no longer consult an encyclopedia in the library but are utilising web-search-motors such as google. Professors have to survey the quality of the information that students gather. Special emphasis was put on the ESARDA website[6] as information source at the end of the course.

Especially in the nuclear safeguards field the information is exuberant and it has been recognised that open sources contain a lot of information. In order to control the impact of the information spread over the web, it is needed to assess its value. Moreover to invest in well-trained future professionals, it is of high relevance to provide the students with that knowledge that they can judge, evaluate and interpret the large amount of information. So the availability of information "approved/ validated" by nuclear specialists will give a reference point to students in a fields were professors and specialists are becoming rare.

4. Perspective with future collaborations

Good course modules require different reviewers and being aware of similar initiatives in Europe (ESARDA), US (PNL, LANL etc.) and in the RF (IPPE etc.). It is of mutual interest to co-operate and to bring together the existing available information of high quality. Specialised institutes might have interest in detailed course modules, which could be added in a second step to an extended course. With the existing highly valuable and detailed information package e.g. on Non Destructive Assay by Los Alamos [6] or on Non-Proliferation Data by Monterey [7], or on specialised NMAC topics of the tripartite seminars in Obninsk [8] a collaboration should enable to establish a common nomenclature and symbol list and to include the links to their appropriate web sites.

An Internet Forum coupled with a safeguards training web site would help thesis and PhD students substantially. Also information on the practical modalities can be included by inserting links to Marie-Curie Fellowships, JRC

[5] https://minerva.Ugent.be/claroline/ssl/login.php

[6] http://www.jrc.cec.eu.int/esarda/

category 20 grants, INTAS trainee positions etc.

Bibliographical References

[1] Dickman, D. A., A Critical Element to Successful Implementation of Future Safeguards Systems, proc. ESARDA May 2004

[2] OECD, Nuclear Education and Training: Cause for Concern? http://www.nea.fr/html/ndd/reports/2000/nea2428-education.pdf, 2000

[3] BNEN: Belgian Nuclear Engineering Network: http://www3.sckcen.be/BNEN

[4] ENEN: European Nuclear Engineering Network: http://www3.sckcen.be/ENEN

[5] Bedke, M. International Safeguards Additional Protocol Implementation in the Department of Energy and National Nuclear Security Administration, http://www.ornl.gov/~websorks/cppr/y2001/misc/117786.pdf

[6] Horley, E. C. (LANL), eLearning and Certification for Non-Destructive Assay Operators and Inspectors, proc. 45th INMM (Orlando) July 2004 with CDROM on NDA Encyclopedia & Training Modules, No. LA-UR-03-5108, Oct. 2003

[7] Center for Nonproliferation Studies (Monterey Institute for International Studies), CDROM on Proliferation Databases including Nuclear Database, No. CNSDATA 4.0, May 2001

[8] Ryazanov, B. four Tripartite Seminars, resp. NMAC for fuel fabrication plants, NMAC for radiochemical plants, Nuclear material measurements and evaluation for physical inventory, Assessment of nuclear materials content and inventory in by-product streams http://www.rmtc.obninsk.ru/seminars/sseminar_eng.html, Obninsk, 1997, 1998, 2000, 2002

	Tuesday, 01.03.05	Wednesday, 02.03.05	Thursday, 03.03.05
8:45	Entrance permission	Entrance permission	Entrance permission
9:00	1. Overview on the treaties: NPT, AP, CTBT, START, ... (A. Poucet)	Solution monitoring and Near Real Time Accountancy (G. Maenhout)	Visit to the PERLA laboratory (P. Peerani)
10:15	coffee break	coffee break	coffee break
10:30	Nuclear fuel cycle (G. Maenhout)	Destructive Analysis: Measurement principles and uncertainty (K. Mayer)	Visit to the TAME laboratory (G. Maenhout)
11:45	Principles and logic of nuclear safeguards (K. van der Meer)	Environmental sampling, particle analysis, nuclear forensics (K. Mayer)	Satellite Monitoring (M. Thornton)
13:00	Lunch	Lunch	Lunch
14:00	Overview of Safeguards Techniques (K. van der Meer)	Non destructive analysis: gamma-spectrometry (P. Peerani)	Visit to the C/S laboratory (V. Sequeira)
15:15	coffee break	coffee break	coffee break
15:30	What is good accountancy from statistical point of view ? (M. Franklin)	Non-Destructive Analysis: neutron-counting (P. Peerani)	Presentation of ESARDA (L. Bril)
16:45	Auditing an accountancy (M. Franklin)	Surveillance and Remote Monitoring, Design Information Verification (J. Goncalves)	Conclusions and Wrap-up
18:00	Closure	Closure	Closure

Fig. 3: Schedule of the first ENEN safeguards course, Ispra, 1-3 March 2005

The ESARDA safeguards course: a state-of-the-art

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Abstract:

In the beginning of 2004 the ESARDA Steering Committee decided to establish an ad hoc Working Group on Modules of Courses.The aim of this Working Group was to create a safeguards course. The course should be at a university level for a target public of nuclear engineers, political scientists, lawyers, etc. Given the diversity of the target public, the course should consist of modules allowing to adapt to the students' background.

The objective of the course is to establish a database of safeguards knowledge that can be used for the education of the future generation of safeguards experts. Given the decline of nuclear energy and the subsequent loss of interest of students for nuclear engineering in general, it was considered necessary to provide the academic world with a tool to raise interest of students for safeguards.

The modular character of the course should allow the professors to use the course for different target publics. From a general overview of safeguards legislation for law students to detailed technical explanation of safeguards measurement techniques for nuclear engineering students; course modules should allow the professor to create a course for a wide range of students.

The content of the course consists of:

- *Safeguards legislation*
  - *International Treaties & Agreements*
  - *National legislation*
- *(Nuclear fuel cycle)*
- *Development of Safeguards Approaches*
  - *From INFCIRC 66 to Additional Protocol*
- *Basic principles safeguards*
  - *Safeguards goals*
  - *Nuclear Material Accountancy*
  - *II, PIV, C/S, CoK, …*
- *Verification techniques*
  - *Sealing, Monitoring, NDA, DA*

This paper describes the state-of-the-art of the ESARDA safeguards course, modules that have been written, partners who have contributed and parties that have shown an interest in the course.

Keywords: safeguards course

1. Introduction.

Although very recently a small revival of nuclear energy is observed, the last decades have shown a decline in the construction of nuclear power plants and a subsequent loss of interest of students for nuclear engineering. This has resulted in a diminuation of the supply of nuclear engineering courses at the european universities. On top of this, even in the booming period of nuclear energy safeguards has never been part of an established nuclear engineering course programme.
For these reasons ESARDA has decided to start an ad hoc Working Group on Modules of Courses with the aim to establish a safeguards course for a broad target public of nuclear engineers, political scientists, lawyers, etc...
This paper will describe the state-of-the-art of this initiative. It will focus merely on training activities, whereas in another contribution to this session the education activities will be reported.

2. Content of the course

The general content of the course as decided by the Working Group consists of the following modules:

- Safeguards legislation
  - International Treaties & Agreements
  - National legislation
- (Nuclear fuel cycle)
- Development of Safeguards Approaches
  - From INFCIRC 66 to Additional Protocol
- Basic principles safeguards
  - Safeguards goals
  - Nuclear Material Accountancy
  - II, PIV, C/S, CoK, …
- Verification techniques
  - Sealing, Monitoring, NDA, DA

As mentioned, the target public for the course is very broad, ranging from engineers via lawyers to political scientists. Therefore not only the technical aspects of safeguards are treated, but also the aspects of legislation, part of the historical background of safeguards and the more general framework of other arms reduction treaties. In view of the diversity of the target public the course should be built up in a modular way in order to be able to adapt it to the target public. This being said, it is considered very useful that the different kinds of public are aware of the aspects of safeguards that fall outside their specialisation.

Based on the above-mentioned content, an ESARDA safeguards course has been given on the premises of JRC Ispra from 1-3 March 2005.

Safeguards legislation – This part discussed the main international treaties and agreements. The Non-Proliferation Treaty is the most widely implemented treaty for nuclear non proliferation, but other treaties like the Comprehensive Test Ban Treaty, Fissile Material Cut-off Treaty and regional treaties like Tlatelolco, Rarotonga and Pelindaba exist as well. Attention was also paid to arms reduction treaties like ABM, SALT, START, SORT, etc. Related treaties are the Biological and Toxine Weapon Convention and the Chemical Weapon Convention for other weapons of mass destruction. National legislation was not discussed, since the public was international.

Nuclear fuel cycle – Included both the front end and back end of the fuel cycle. Starting with mining and an overview of the geographical spread of natural uranium, extraction and conversion processes were discussed. Enrichment by diffusion, centrifuge or lasers was discussed in detail. Several conversion routes for fuel fabrication were shown. The last part of the front end of the fuel cycle is the nuclear reactor. Several types exist like the Light Water Reactor, Fast Reactor, Gas Cooled Reactor, WWER, RBMK, ... The chemistry of reprocessing spent fuel was detailed. The options for waste disposal were shown. Finally alternative technologies like transmutation were discussed to provide the students with the insight that nuclear research still has a future.

Development of safeguards approaches – This section dealt with the historical development of the safeguards approaches that have been used by the IAEA. The first system INFCIRC/26 and its successor INFCIRC/66 were limited to specific nuclear material and nuclear facilities, included some non-nuclear material. The signature of the Non-Proliferation Treaty changed this situation. Full-scope safeguards as described in INFCIRC/153 was implemented gradually and today only three countries have a safeguards regime under INFCIRC/66, notably Israel, Pakistan and India. The discovery of a hidden weapon programme in Iraq after the First Gulf War initiated the development of strategies to verify the absence of undeclared activities. This culminated in the Additional Protocol, INFCIRC/540.

Basic principles of safeguards – Items that are discussed in this part are the political and technical scope of safeguards and the limitations of the system. The safeguards principles for declared material under INFCIRC/153, consisting of the starting point of safeguards at material suitable for enrichment, safeguards techniques with the basis of nuclear material accountancy, supplemented by C/S measures. Verification techniques like non-destructive and destructive measurements. Some definitions like significant quantity, timeliness goal, detection probabilities and material categories. Diversion strategies and counter measures by inspectors, several types of inspections. Safeguards principles under the Additional Protocol for undeclared activities and measures like open source information, satellite monitoring and environmental sampling.

The statistics of nuclear material accountancy – The basics of nuclear material accountancy in a Material Balance Area were explained. What is a beginning inventory, shipment, receipt, ending inventory, transfer, nuclear transformation, material processing and how is the Material Unaccounted For (MUF) defined. What is the role of measurement errors in MUF and what is acceptable and what is not. What strategies can an operator follow to divert a significant quantity and what is the best strategy for an inspector to verify a number of items with different inspection methods. How can the false alarm probability and non-detection probability be reduced. What is the optimum combination of the two. Finally a practical example of attribute-variable sampling was given for two and three inspection methods with different measurement uncertainties.

Solution monitoring and Near Real Time Accountancy – After a brief introduction of the problem of monitoring solutions during the reprocessing, details were given on the technologies and methodologies used for verifying the nuclear material accountancy in the reprocessing facility near real time. Special aspects such as status of health registration and syntactic pattern recognition for batch process control are addressed to illustrate the comprehensiveness between a MUF declaration and the problem localization by means of nuclear material monitoring.

Destructive analysis – Criteria for the application of Destructive Analysis were discussed, like costs, analysis speed, selectivity and accuracy. The philosophy of on-site laboratories in LaHague and Sellafield was explained. The destructive analytical techniques were discussed in detail, like Hybrid K-edge and Isotope Dilution Mass Spectrometry with Thermal Ionisation Mass Spectrometer. Some figures of the present Euratom safeguards efforts were presented.

Environmental sampling, particle analysis, nuclear forensics – A relatively new area in safeguards analytical techniques was presented in this session. From environmental sampling in the framework of the Additional Protocol swipe samples are taken from installations to look for undeclared activities. Particle analysis is applied to these samples to look for traces of plutonium or highly enriched uranium. Nuclear forensics is applied in the framework of the battle against illicit trafficking of nuclear material. Suspect items caught by costums are analysed by several different techniques to determine the origin. Gamma-spectrometry, SEM, EDX, mass spectrometry, classical chemistry are used to determine as many as possible parameters and to compare these with a database. Some examples were discussed.

Non-destructive analysis: gamma-spectrometry – The principles of photon detection by various detector types were explained. Scintillators like NaI are efficient but the resolution is rather poor, while semiconductors like Ge are less efficient but have a high resolution with a strong capability of isotope characterisation. CdZnTe are less efficient than Ge but do not need to be cooled down. The electronics for signal processing was discussed by explaining the function of preamplifier, amplifier and MultiChannel Analyser. Pile Up Rejection, Pole Zero Adaptation, dead time and spectrum stabilisation were explained. The application of gamma-spectrometry to measure the isotopic composition of uranium and plutonium was shown.

Non-destructive analysis: neutron counting – Neutron sources, basic principles of neutron interactions with matter (moderators, absorbers, fissile material, cross sections), neutron detection principles. Neutron detectors based on ^{3}He, BF_3, ^{235}U. Principles of coincidence counting compared to single neutron counting, passive and active methods. Shift register, active well coincidence counter, Neutron coincidence collar, High-level neutron coincidence counter were subject discussed in this session.

Surveillance and Remote Monitoring, Design Information Verification – The principles of Containment and Surveillance, especially verification in unattended mode were dealt with. The issue of design information verification was addressed to illustrate the importance of laser techniques and image processing. Many interesting details were given concerning remote sensing techniques by satellite imagery.

Various visits of JRC laboratories – PERLA, TAME, satellite monitoring, C/S, ESARDA presentation

The institutes that contributed to this course were JRC Ispra, Italy, ITU Karlsruhe, Germany and SCK•CEN Mol, Belgium.
Most contributions were given with the help of transparancies. For a few sessions written notes were available, e.g. Safeguards Legislation, Nuclear Fuel Cycle, Statistics of Nuclear Material Accountancy.

The course was attended by eigtheen participants with various backgrounds, although most of them were engineers. Four students were from the BNEN (Belgian Nuclear higher Education Network) nuclear engineering programme, a joint programme of several Belgian universities and SCK•CEN. Three students were from the Department of Physics Engineering of the University of Ghent, three from the Mechanical Engineering Department of the University of Glasgow, one from the Economy Department of the University of Geneva and three interns from JRC.

3. Improvements to be made

Being the first of its kind in the ESARDA context, there is certainly room for improvement of the course.
A first handicap is the (partial) lack of documentation. The content of the majority of the sessions still has to be written down in notes so the students can rely on text rather than only on a few transparancies.
Second the content of the course is not as comprehensive as desired. The latest developments in safeguards like the Additional Protocol have been touched but not extensively explained. Integrated Safeguards should be dealt with in a complete session. Safeguards approaches for specific nuclear facilities like the Hexapartite Safeguards Project for centrifuge enrichment plants and the developments in the safeguards approach for geological repositories are of interest.
On the contrary, it was felt that one elaborated too long on the development of the safeguards approaches and that one could spent a little more time on the safeguards principles.
The visits to the laboratories should be spread over the whole duration of the course and not concentrated on one day.

4. Conclusions

The aim of the Working Group to have a comprehensive safeguards course with both transparancies and written documentation at the start of the academic year 2005-2006 will probably not be met completely. However, a lot of presentation material does exist now and can be supplemented with written documentation.
Parts of the course will be available for European universities for the academic year 2005-2006.

Quite unexpectedly ESARDA succeeded to give already a safeguards course during 2004-2005 to a reasonably sized group of students. The feedback of both teachers and students was on the whole positive, so that the course is intended to be given again next year.

A Sustainable Approach for Developing Next Generation Safeguards Instrumentation

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Abstract:

The challenges are manifold to provide the International Atomic Energy Agency (IAEA) with the instrumentation needed to support safeguards inspectors in their mission to verify compliance with states' non-proliferation commitments. The IAEA needs custom-designed, high-quality equipment that has to be supplied and supported for at least ten years but is ordered infrequently and only in small quantities. Moreover, manufacturers have to cope with rapid obsoleteness of electronic components. This makes it difficult for research, development, and manufacturing entities, as well as Member States Support Programs (MSSPs) to handle the lifecycle management of such equipment. Furthermore, the process to identify the need for certain safeguards instrumentation, to define its requirements, to initiate its development, and to finalize its design is often lengthy and cumbersome. In addition, once development is completed, more advanced technologies are already available. To overcome these problems it is essential to pool the experiences of developers, manufacturers, MSSPs, and the inspectorates, gathered over the last three decades of development of safeguards equipment. The following paper will identify all essential partners that support the IAEA in their safeguards mission and will list their contributions to successful instrumentation development. Next, the paper will give insight on how these partners need to combine their efforts to provide a sustainable approach for equipment development and supply, as well as for ongoing factory support. The development of the Next Generation Surveillance System (NGSS) will serve to illustrate how some essential steps of such a partnership approach have already been implemented. Recommendations for further broadening the partnership will conclude the paper.

Keywords: sustainability, safeguards, surveillance, instrumentation

1. Introduction

In its mission to verify the compliance of Non-proliferation Treaty (NPT) Member States with their Safeguards agreements, the International Atomic Energy Agency (IAEA) relies on a tight and strictly regulated inspection regime. IAEA inspectors make use of instrumentation that provides information from which it can be concluded that no nuclear materials have been diverted from civil applications in the absence of inspectors. Since the first application of Safeguards in the 1961[1], emerging technologies have been applied to strengthen the IAEA's Safeguards systems by increasing synergies between inspection efforts and automated control. Digital surveillance and other Safeguards instrumentation is an integral part of the IAEA treaty enforcement strategy at the beginning of the 21st century.

Reliance on Safeguards equipment inherently carries significant downsides which, if not managed properly, threaten to offset the increased assurance that Member States comply with their non-proliferation agreements. On the one hand, the IAEA finds itself locked in a development race, since as new Safeguards technologies become available so do enhanced means to defeat them. This issue is mitigated by the fact that Safeguards instrumentation is still considered an additional proliferation detection tool, which merely supports the personal visits of IAEA inspectors to nuclear facilities. There are, however, other, more mundane issues that have a more significant impact on the usage of instrumentation related to the unique position of the IAEA as an international treaty enforcement agency. These problems are encountered during the procurement, development, funding, installation, and maintenance of Safeguards equipment.

The following paper will outline the complicated problems and needs of the IAEA in its commitment to strengthen its detection capabilities with Safeguards instrumentation. It will describe the Agency's various partners and identify their contributions to successful instrumentation development as well as problems encountered in partnering with the IAEA. Next, it will give insight on how all partners have to pool and coordinate their efforts to be able to provide the Agency with a sustainable management approach and to support Safeguards instrumentation over the complete lifecycle. The Next Generation Surveillance System (NGSS) will serve as an example how essential steps to such a partnership have been recently realized. Further recommendations on broadening this partnership will conclude the paper.

2. The Unique Problems Related to IAEA Safeguards Instruments

The IAEA investigates all suitable markets to minimize the procurements costs of Safeguards instrumentation. The least costly alternatives are usually available in the consumer electronics market, where items are offered inexpensively and competitively in volumes in the millions. Compared to this, markets for industrial electronic components, measurement technologies, or surveillance and monitoring systems are relatively small with customized production volumes in the low thousands. Industrial electronic component markets are more expensive but are more likely to offer solutions suitable for Safeguards. Parts stemming from all of these markets include, for example, CCD cameras, digital storage media, and personal computers. In the design of a Safeguards system, commercially off-the-shelf (COTS) components usually have to be complemented by custom designed parts to meet the specific needs inherent in the application of Safeguards instrumentation such as include tamper indication, data authentication and encryption, and high reliability. The same complementary process applies to software products; usually, executable programs have to be adapted for Safeguards specific applications or written as customized programs.

The IAEA has neither the personnel nor the resources to accomplish this custom development work in-house. Thus, the Agency has to contract these projects to commercial entities and research and development institutions. Funding is provided by Support Programs that assist the IAEA with extrabudgetary funding for development and procurement of Safeguards instrumentation. Due to the Safeguards-specific user requirements, the development of both hardware and software is time

[1] The first installation safeguarded by the IAEA was the Japanese JRR-3 research reactor, with a Safeguards system approved by the IAEA Board of Governors in 1959 and implemented in 1961.

consuming and expensive. Manufacturing of such systems must also be accomplished by external commercial entities, as orders are small in volume, are placed irregularly, and require very high quality control. The total number of systems needed varies from dozens to a few hundred every year or even in total. For some developments, only a few systems are required, making it impossible to realize significant Economies of Scale.

Safeguards-specific instruments have a low potential of attracting customers outside their niche market.[2] This puts the IAEA into a monopsonistic position, as the only buyer of Safeguards systems. On the other hand, the low volumes needed support only one manufacturer per system, adding a monopolistic provider to the already complicated Safeguards supply environment. Whereas the IAEA issues Basic Supply Agreements (BSAs) that state the number of systems they intend to purchase over a certain period of time at a fixed price, BSAs do not constitute a commitment to really purchase systems. This makes it impossible for manufacturers to anticipate and prepare for incoming orders without bearing the financial risk of stockpiling parts without receiving an order.

Further complication stems from the looming threat of crucial equipment parts becoming obsolete in the fast moving electronics market. Once a newly developed design has been authorized by the IAEA for inspection use, market-enforced technical changes require a lengthy and cumbersome process, drain Agency resources, and re-occur on a regular basis.[3]

Even though the situation faced by the IAEA seems so complicated and cost-prohibitive as to prohibit employing instrumentation at all, the problems can be mitigated and even solved if properly managed. Before appropriate solutions can be offered, however, it is necessary to examine more closely the individual parties involved in the development, manufacturing, and support of Safeguards equipment.

3. Safeguards Instrumentation Parties

3.1 IAEA

As the central customer for Safeguards instrumentation, the Agency is responsible for procuring the development of appropriate technologies. Unfortunately, this must be accomplished with a strained budget. Because not enough personnel are available in-house to develop and build equipment, detailed user requirements must be communicated to potential partners to allow them to develop solutions in accordance with concrete IAEA needs. The identification of user requirements tends to be a complicated process as it involves multiple sections (e.g. technical support, operations, and concept and planning). Developments in the nuclear sector such as new reactor types[4] or new non-proliferation policies[5] sometimes call for new Safeguards approaches which have no reference to past projects, require innovative initiatives, and are impossible to test because the facilities they are to safeguard do not yet exist.

Without the input of commercial partners or research institutions specializing in the relevant area of the new approach it would be virtually impossible for the IAEA to develop suitable user requirements on its own. However, this bears with it inherent problems, in that a partner may attempt to influence the requirements development to favor an approach that is not necessarily the solution favored by the IAEA', but is appealing to the partner because it is easier to accomplish, cheaper, or inaccessible to possible competition.

One way to cope with these problems is for the IAEA to pool as much expertise as possible in-house to proof and criticize all second party input. An alternative method is to draw upon all available outside resources and expertise to draw an objective picture of what is needed. As an international agency supported both by its Member States in general and Member States Support Programs (MSSPs) in

[2] Some customers include the European Commission (EC), the Brazilian-Argentine Agency for Accounting and Control of Nuclear Materials (ABACC), and a few others, all with volumes lower than the IAEA.
[3] For example, Central Computer of the Digital Multi-channel Optical Surveillance System (DMOS) used by the IAEA is known to becoming obsolete about every six months.
[4] Such as the Pebble Bed Modular Reactor (PBMR) development or the reactor types investigated by the Generation IV forum.
[5] Especially the Additional Protocol INFCIRC /540 (corrected) and Integrated Safeguards.

particular, the Agency has been very successful in employing both methods, attracting international experts to share knowledge (often at minimal expense) and to work for the IAEA on staff.

Another problem with the potential to significantly hamper Agency efforts to obtain Safeguards instrumentation lies in the procurement regulations of the United Nations. New development projects exceeding a certain expected cost volume have to be issued as open tenders, inviting all interested parties into an open bid process. In broad consumer markets this process has the advantage of attracting comparable offers at competitive prices. This is not necessarily the case in the Safeguards area. The personal experience that specific companies and research institutions have in Safeguards as a whole and with IAEA user requirements development is essential to providing a project schedule and delivery plan at a fair price. In addition, devotion to the non-proliferation of nuclear weapons on the part of IAEA partners is an important aspect which should not be overlooked.

However, under current UN procurement rules, it is possible for a new company with a general interest in the Safeguards market to bid low on and win a tender, misjudging the boundary conditions of Safeguards. Later, after the effort needed has become obvious and has been recalculated there is a risk that the cheaper alternative becomes more expensive in the long term, as a newly awarded partner must slowly acquire the experience which an established partner would have calculated into its initial offer. In the worst case scenario, a new partner withdraws from Safeguards after failing in its efforts while in the meantime the established partner has dissolved or moved on to other markets because there was no room left in Safeguards for more than one company. The Agency would have expended time and resources only to be left without any partner available or willing to assist in the development of Safeguards instrumentation.

3.2 Member States Support Programs

Several MSSPs[6] provide extrabudgetary assistance to the IAEA through funding of development projects, equipment purchases, staff experts, and international exchange of expertise. Other than the regular contributions all Member States pass to the Agency every year, these extrabudgetary resources are subject to the specific funding regulations of each individual MSSP. The United States Support Program for IAEA Safeguards (USSP), for example, can only provide funds to the IAEA if they are used for purchases of equipment or services in the US. This regulation is a means for the US government to support both the international Safeguards community and its domestic suppliers. The German Support Program (GERSP)[7] was established for similar reasons: strengthening IAEA and EURATOM Safeguards, preserving German expertise in the Safeguards area, communicating changes in the German nuclear program to the IAEA, and supporting the German industry.

It must be stressed that MSSP assistance is essential to relieving the IAEA's strained regular budget and in providing instrumentation and expertise for comprehensive treaty enforcement capabilities. Due to the nature of the MSSPs, however, there are a few issues impacting the IAEA's ability to make optimal use of the offered assistance. Since part of the MSSP mission is to support the economy of its country of origin, many programs are interested in not only having local companies and research institutions develop new Safeguards instrumentation but also in manufacturing the systems for the IAEA after development has been completed. The developer usually has an advantage of experience and thus a good chance of being selected as manufacturer (assuming he has the capabilities to do so). MSSPs therefore promote local developers for Safeguards instrumentation to secure not only funding for development, but also BSAs during later production.

At times multiple MSSPs have initiated the development of the same Safeguards instrumentation in parallel.[8] Since Member States must be treated equally and no effort to support the Agency with instrumentation should go unrewarded, the IAEA Safeguards Department had no choice but to evaluate and test all candidates. Generally, there is only enough demand for a single instrument in

[6] States and organizations representing groups of states having formal support programs: Argentina, Australia, Belgium, Canada, Czech Republic, EC, Finland, France, Germany, Hungary, Japan, Republic of Korea, Netherlands, Russian Federation, South Africa, Sweden, UK, and US.

[7] The complete title of GERSP: Joint Programme on the Technical Development and Further Improvement of IAEA Safeguards between the Government of the Federal Republic of Germany and the International Atomic Energy Agency.

[8] For example, three MSSPs initiated the development of a seal to replace the fiber optic VACOSS seal: the GERSP (EOSS), the French Support Program (IRIS), and the USSP (VACOSS 5E).

each respective category, drawing IAEA resources into a lengthy selection process. Further exacerbating the problem, the efforts of MSSPs that did not get selected are wasted because applicability of Safeguards instrumentation is limited in other markets.

Another issue encountered in the past lies with the intellectual property rights (IPRs) generated in projects funded by MSSPs. For example, the German government, as part of its mission to preserve Safeguards expertise, requires that all IPRs generated through funding of the GERSP are vested with the German government, while the development results are provided to the IAEA free of charge for worldwide Safeguards applications. Although Germany guarantees that there will always be a partner for the IAEA to procure Safeguards instruments governed by the GERSP and that the prices will remain “fair”, in its mission to support the national economy, the GERSP also has an interest that licensees are German companies. With regard to supply assurance, during the developer selection process the GERSP requests a commitment on the part of the developer that they will also be the future equipment supplier However, this prevents the IAEA from using available global resources and competition to acquire instrumentation at a competitive price.

This issue is mitigated by the nature of the Safeguards market. While competitive procurement, as mentioned above, is not necessarily helpful for instrumentation development, it might make sense for the production of equipment. However, at that point, most of the funding has already been spent, and the differences in production prices tend not to differ a great deal. In addition, as will be explained below, the number of companies willing and, more importantly, capable of supporting Safeguards instrumentation over the complete lifecycle is limited. Thus, protecting the developer by governing the IPR and by issuing licenses only to qualified manufacturers, the GERSP preserves Safeguards expertise, supports domestic economies, and still enables the IAEA to procure instrumentation at reasonable prices. Also, it should be noted that if the IAEA were to select a manufacturer other than the developer, it would be necessary to repeat equipment authorization; a process that can take up to a year or even longer.

3.3 Commercial Entities and Research and Development Institutions

The IAEA and the MSSPs have a large number of both commercial and governmental cooperation partners which face differing obstacles and offer different solutions when dealing with the Safeguards market.

3.3.1 Large Companies

Large commercial entities have significant advantages in both the development and manufacturing of electronic instrumentation. In-house capabilities in research and development, tight quality control and documentation requirements, internal auditing, as well as financial security are just a few of the strengths that make large companies attractive partners for the IAEA. The downside, however, is that such companies are generally focused on markets that are magnitudes larger than the Safeguards market in production volumes. In-house processes are focused on automated large scale production, learning effects, and Economies of Scale. Experience has shown that, while large companies might be interested in supporting the research and development of Safeguards instruments, they are not interested in the small Safeguards market with its high production cost due to high quality standards and low order volumes which do not allow for automated production.

There is a converse danger that large companies overestimate the potential for Safeguards equipment for other markets, decide to enter into a bidding process, and then withdraw at some point when the actual market potential becomes apparent. Such involvement can cause significant delays in the initiation and conduct of research and development projects and can drain both IAEA and MSSP resources. Some examples are given:

A large German company serving the consumer electronics and security markets was asked to propose the development of an authentication method of digital surveillance images. Initial negotiations took months and did not result in a proposal. The company had problems with the highly specific user requirements, and the potential markets were considered too small. Also, communication with the IAEA and the Commission of European Communities (CEC) was slow and prone to misunderstanding. The company eventually lost interest, causing significant delays in commencing the project.

Another German company successfully developed the active VACOSS Sealing System but could not provide the required level of quality during the streamlined manufacture of the seals even though there was a high-standard quality control system in place. After two production runs of 100 seals each, the IAEA removed this company from its suppliers list. Evidently, this company was not interested in supplying the Safeguards market. This resulted in the GERSP requesting an a priori commitment to both develop and manufacture the development product.

A large Japanese company developed a few prototypes of an automated Cobra Sealing System reader that did not work reliably enough for Safeguards applications. The anticipated market was considered too small and the additional development effort needed too large, and the company withdrew their commitment to manufacture the readers. Even worse, once interest in the project had dissipated, the company was either unable or unwilling to provide the to-build documentation of the reader to the IAEA, making it impossible for the Agency to look for another manufacturer.

3.3.2 Mid-sized Companies

Generally, mid-sized companies are better partners for the development and manufacture of Safeguards specific instruments. The total anticipated market volume is more appealing to such partners and negotiations and user requirements finalization can take place on a much more intimate level. Certain mid-sized companies have evolved to be reliable partners to the IAEA over the past decades, building the backbone of the supply of Safeguards instrumentation.

Unfortunately, there are problems in partnering with mid-sized companies as well: they have the capability but also the necessity to focus on a certain field of expertise to remain competitive in a specialized area. This can result in a company misinterpreting the potential and specific needs for Safeguards instrumentation. For example, a German company specializing in the development and manufacture of perimeter security equipment discontinued a project for the authentication of analog video signals after it became obvious that the Safeguards market was not as similar to the security market as it first appeared. The company realized that there was little applicability for the technology to be developed in their core capacities and lost interest.

Mid-sized companies that support Safeguards thus must be committed to Safeguards and must consider it their core capability. Two of the authors represent one such company. Initially, this company received a license from the GERSP for the manufacture of the VACOSS seal after the German developer terminated production. Cooperation with the GERSP and the IAEA was extended by acquiring further licenses for the production of the DCM 14 based Safeguards surveillance system, working closely with a small German company. In cooperation with the USSP, the IAEA, and the CEC the company also initiated and successfully completed its own Safeguards specific developments (such as the General Advanced Review Software (GARS) for the review of Safeguards surveillance data). In addition, the company successfully worked with research and development institutions in the provision of Safeguards instrumentation, mitigating some problems related to such IAEA partners.

3.3.3 Small Companies

At first glance, the cooperation of small companies with the IAEA seems to be the ideal combination from a project management point of view. Small partners have the ability to fully concentrate on the IAEA as a customer and conduct the highly specified customized developments with their full attention and resources. Inherent in the structure of such small partners is the capacity to build a developer-customer relationship on a very personal level.

On the negative side, small companies are very dependent on their specialized niche market. Personal experience gained during previous projects is highly specific and not necessarily applicable to other markets; thus, should the niche market change, or should competition try to enter it, it is likely that a small company will be forced to either withdraw from Safeguards and seek other markets, or dissolve, since usually the market only supports one supplier for each respective Safeguards system. This applies not only to development projects but also to instrumentation production. Small companies usually have little or no other customers and are thus heavily dependent on a constant stream of orders to continue operating. Traditionally, the IAEA has ordered equipment irregularly through BSAs, making it difficult for a small partner to survive just on the basis of Safeguards.

There are a few successful small companies that support the Safeguards market, one of which is the above mentioned developer of the DCM 14 technique. However, over the past 15 years of supplying development services as well as instrumentation to the IAEA, this company has experienced exactly the problems described. To avoid bankruptcy due to lack of orders, the company partnered with a mid-sized US company, with the financial strength to spread IAEA orders over a period of time, thus providing DNE with a constant line of income. This small company also enjoys the strong commitment of the GERSP to conserve its Safeguards-specific experience.

3.3.4 Research and Development Institutions

Like small companies, government funded research and Development (R&D) also appear at first glance to be ideal partners for the IAEA in the development of highly specialized and complex Safeguards instrumentation. R&D facilities expand knowledge and science to new levels on a regular basis. Resources and expertise are abundant at such facilities, and there are few other institutions that can so fully grasp both the IAEA mission and the related technical requirements. Nevertheless, the IAEA's cooperation with research laboratories has been overshadowed by difficulties.

First of all, while the understanding of technical requirements is the most important issue related to the development of new Safeguards instrumentation, it is only a part of the whole. Maintainability in the field, operability by IAEA inspectors, ruggedness against harsh environmental conditions, and cost-effectiveness in the parts used are other factors that need to be considered when designing Safeguards solutions. While the R&D facilities excel at understanding a technical problem and providing scientifically sound solutions, they often lack the expertise to manufacture, install, operate, and maintain equipment in the field. Technologies have to be passed on for commercialization by an R&D facility after reaching prototype level. This is the stage where problems become apparent.

Commercial partners often find it difficult to adapt prototype technology to a commercially acceptable product with a streamlined manufacturing process. This transition is further complicated by the strict requirements of Safeguards instrumentation. Operating with a lack of commercial pressure, R&D institutions tend to pursue the most scientifically sound solution, which is not necessarily the one easiest to build, maintain, or support. Also, once a technology has been developed to the prototype stage, the scientists involved are required to move on to other projects and are therefore unavailable to assist in commercialization to the fullest extent.

In this respect it is essential that both the R&D institution and the commercial entity work together as early as possible to get an understanding of what a scientifically sound and commercially feasible solution should look like. This hardly ever takes place, though, because the IAEA, also in accordance with procurement regulations, has to wait for the development to be completed before issuing the commercialization and production of the instrument as an open tender. This is further complicated in that the IAEA usually does not contract the R&D facilities directly but with the respective MSSP as intermediary. This sometimes makes it difficult for the IAEA to judge the progress of a project and what can be done to facilitate an easier production of the instrumentation after the development has been completed.

4. Player Management

While the previous paragraphs seem to draw a dark picture of all the problems accompanying the facilitation of Safeguards instrumentation development and production, most of the issues encountered can be solved with a single attitude: Commitment to Safeguards. If all parties understand the problems related to the unique realities of Non-proliferation Treaty enforcement, all individual strengths can be enhanced while weaknesses can be negated.

It seems contradictory to suggest the IAEA to be committed to Safeguards, because in more than technical terms the IAEA *is* Safeguards. But commitment in this regard means commitment to the Safeguards market. The IAEA needs to look upon itself as a complex customer with a wide range of (often conflicting) user requirements that might change over the lifetime of a system development. It is also a difficult customer because it shows little or no commitment to the supplier other than BSAs, which merely fix the price for a system but do not offer enough security for a company's proper risk

management. Low order volumes and irregular order practice further aggravate this problem. The IAEA has been very good at communicating an expected order outlook to its partners, but this process is not institutionalized and relies on the supplier's expertise to anticipate the proper need for each fiscal year. An official Safeguards needs meeting, conducted on an annual basis and involving all affected parties could help suppliers build up confidence in expected order volumes and help them achieve Economies of Scale, help prepare for parts obsolescence, and provide a Safeguards strategy to provide identical Safeguards instrumentation for the complete lifecycle of equipment. This would also provide the IAEA with more supply assurance.

The mission of MSSPs is not only commitment to Safeguards but also to other objectives such as those stated above. However, these different commitments can be combined if a comprehensive effort is undertaken to strengthen Safeguards as a whole. Instead of pursuing multiple development efforts in the same area, each MSSP should evaluate the strengths of its domestic commercial partners to the IAEA and research institutions and decide how these strengths can be applied to support this comprehensive approach. Also, it is essential for MSSPs to identify domestic supporters to the IAEA and evaluate how to best keep their contribution capacities and experience with Safeguards available for future application. If possible, MSSPs should provide sufficient funding to prevent obsolescence problems if last-time buy opportunities can be identified, and threatened parts can be stocked for the remainder of the Safeguards instruments' lifecycle.

Commercial companies have to be fully committed to the IAEA as a customer. Unique and difficult user requirements and the Agency as a bureaucratic giant pose obstacles that will test the patience of each partner, but the goal, the provision of Safeguards equipment, should always be foremost. Similar to the MSSPs, different levels of expertise (e.g., development, production, and commercialization) should combine their efforts where appropriate to mitigate the impact of problems on a single partner and smooth the transition from development to the actual fielding of Safeguards system.

R&D institutions must acknowledge the challenges technologies encounter once they leave the laboratory. Early involvement of commercial partners during the design phase of technologies can lay the ground for easier fieldability of the finalized product. In this respect, R&D installations not only could but should be part of the cooperation efforts that private partners should undertake. Combining expertise in all areas of system development, supported by the MSSPs with close cooperation with the IAEA will make Safeguards systems available more quickly, in accordance with user requirements, sound funding plans, and a lifecycle management in place for the complete lifetime of the system.

5. Conclusion and Recommendations

The IAEA has been very active in pursuing cooperative approaches for the development of new Safeguards systems. For the development of the Next Generation Surveillance System (NGSS), the replacement for the DCM 14 based surveillance system, the Agency pooled international expertise from both public and private partners to identify the needs and requirements of NGSS. A cooperation of two established suppliers of Safeguards was selected with both parties sharing responsibilities in their respective core capabilities. NGSS is funded by both the GERSP and the USSP, a comprehensive effort unprecedented in the history of the development of Safeguards instrumentation. A team of representatives of all involved parties, with communication channels open at all times, will govern the development of NGSS to make it a successful replacement for the next 15 to 20 years.

There is, however, room for further improvement. While the IAEA expressed its intention to begin a development project in early 2003, the definition of user requirements for NGSS was not initiated until October 2003 with a meeting of international experts. It took until April 2004 to initiate the bidding process. This open bidding process was lengthy, involved an extensive selection process, and was finally concluded in March 2005. Hopefully, the experiences and obstacles encountered in the initiation of NGSS can be applied towards future development projects to make it easier for the best available partners to join their efforts towards Safeguards development.

Session 15

NDA quantitative measurements

A Portable Neutron Counter for Field Surveys and Assay

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Abstract:

A portable handheld neutron counter (PHNC) has been developed to support ad hoc field inspections requiring performance under non-laboratory conditions. The system consists of a pair of ^{3}He-based slab counters coupled to a portable, battery operated, electronics unit. The entire assembly fits into a small carrying case and weighs approximately 10 kg. Operation may be accomplished with the detectors either inside the carrying case or assembled in an arbitrary configuration outside the carrying case. Multiple cases can be used to accommodate a variety of sample configurations encountered in the field. Singles mode counting enables area surveys for source search or radiation direction characterization. Directional sensitivity and background correction may be accomplished through proper configuration of the independent slab detectors and use of the dual channel capability in the hand held electronics. Coincidence counting is facilitated by a short die-away time and high efficiency that allows for traditional quantification of small samples. Given the flexible nature of counting geometry, computer models are used as an integral part of the data analysis. A small micro computer is incorporated into the unit allowing real time analysis in a portable configuration. In this paper, we present the detector and electronics design, results of initial laboratory tests of the system, and compare to Monte Carlo simulations.

Keywords: neutron, portable, detector, assay, survey

1. Introduction

Expansion of inspection activities by the International Atomic Energy Agency (IAEA) as a result of enhanced safeguards measures such as implementation of the Additional Protocol and Complimentary Access are posing new measurement challenges requiring new detector systems. In particular, a portable neutron detection system that is field-configurable to adapt to a variety of measurement conditions would facilitate large area surveys, identification of plutonium-bearing materials, and quantitative verification in a single unit. We have addressed this need with the development of the Portable Handheld Neutron Counter (PHNC), consisting of a pair of ^{3}He detector slabs housed in a small briefcase, complete with battery operated coincidence electronics and micro-computer. The entire unit weighs approximately 10 kg and is easily carried and configured in the field.

The PHNC can be operated in multiple modes. Large area surveys can be accomplished by counting singles, with the detectors remaining in the briefcase or positioned in such a manner as to provide directional sensitivity. Coincidence counting can provide verification measurement capability through field configuration of the pair of ^{3}He detector slabs or, for larger samples multiple slab pairs, in a geometry with relatively high counting efficiency. Data interpretation is

aided by the use of a detailed MCNPX [1] model, which enables normalization of the arbitrary configurations used in the field to a standard one.

2. Hardware Description

The core of the system is a ^{3}He-based neutron detector unit developed with PDT [2], with overall dimensions of 18 cm (width) x 25 cm (height) x 7.6 cm. Each slab is made of high-density polyethylene and is 23 cm tall. Four ^{3}He tubes, with active length of 18 cm and 10 atm pressure, are evenly spaced in the high-density polyethylene. The ^{3}He tubes connect to a low-profile aluminum junction box, adding only 2.5 cm to the overall height. The only external power required to operate unit is +5 V. High voltage (typically 1700 V) to bias the ^{3}He tubes is provided onboard from the low voltage power. Each slab weighs approximately 4 kg. The individual detector unit can be operated alone or combined with others in an arbitrary fashion. When used together, the output signal can be summed prior to input to the shift register.

A handheld shift register is being developed to provide the external +5 V power to the detector slabs and to collect data. The system has two channels, and can provide two separate high and low voltage outputs. In the current application, only the low voltage power supply is used and the output of all detectors is summed and processed by a single channel in the electronics unit. This provides a capability for traditional thermal neutron counting [3] techniques to the applied. The overall dimensions are 12 cm (width) x 29 cm (height) x 10 cm, and the unit weighs less than 1 kg. Figure 1 shows a pair of detector slabs and the handheld electronics unit assembled in the briefcase.

Figure 1. A pair of detector slabs and handheld electronics unit assembled in a briefcase.

3. Laboratory Testing

Initial testing of the detector system was done in the configuration shown in Fig. 2. Four detector slabs were placed corner-to-corner to form a collar-type arrangement. In addition, a 5 cm thick polyethylene slab was positioned underneath to provide a reflecting geometry more or less independent of the actual location of the detector slabs. The signal from each detector was combined to form a single output which was then input into an AMSR, and subsequently to a collect computer. All data was collected using the INCC code [4].

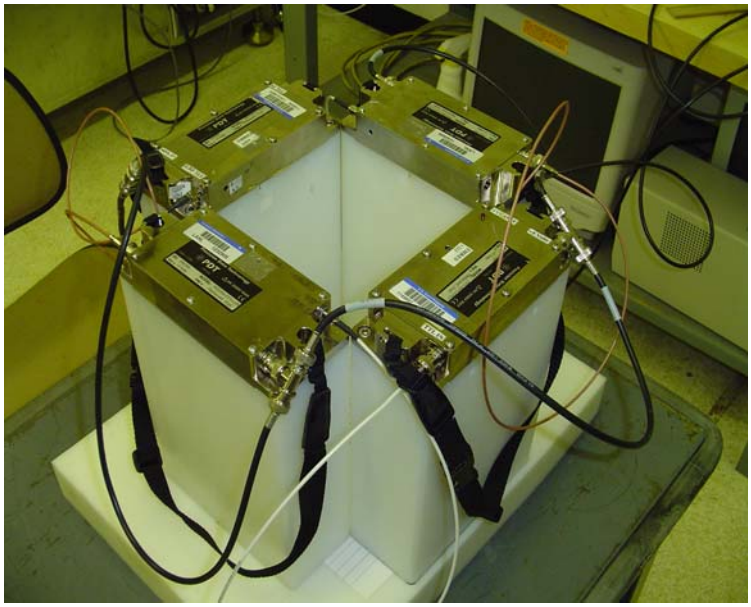


Figure 2. Initial detector set-up (standard configuration).

Characterization of the system was carried out using a ^{252}Cf source. Data was collected for 300s. Overall system efficiency was measured with the ^{252}Cf source placed in the center of the active volume of the ^{3}He tubes. In addition to system efficiency, individual channels were obtained to document the degree of balance between the slabs and response profiles were measured in all three spatial dimensions. Singles and Doubles were collected at each of the positions.

The configuration shown in Fig. 2 is denoted as the standard configuration for the purposes of comparing the system response in other configurations. The flexibility of the PHNC allows for configuration of the detectors to accommodate sample sizes encountered in the field. Thus, one can optimize the efficiency of the counting geometry and then normalize the response to the standard configuration. A critical aspect of this approach is the normalization process, which is aided by the use of a detailed MCNPX model.

4. MCNPX Model

Monte Carlo modeling of neutron coincidence counters has matured in the last several years to the point of incorporation of features that specifically tailored for that purpose are included in the MCNPX code. Capture tallies on ^{3}He can be setup now in the code itself so that the detector response (Singles, Doubles, Triples, etc.) may be simulated directly, without the use of the point model. In addition, fission is now treated in a detailed fashion with respect to the multiplicity

distribution. A current overview of new features of the MCNPX code and its application to safeguards is presented in a companion paper [5].

The standard configuration, as shown in Fig. 2, was modeled using the MCNPX code. Details of the model include full representation of the ^{3}He detectors, including dead areas, the aluminum junction box, high-density polyethylene moderator and reflector. Details of the features of the room where the measurements were conducted were not included as it was assumed that the detector was far enough from wall and floors to make little difference in the response. Capture tallies on ^{3}He were setup for direct comparison to measurement data. The 252Cf source was modeled as a point source and positioned as was done in the measurements. Details of the source encapsulation were not included, nor was the positioning rod. Figure 3 shows the MCNPX model of the standard configuration.

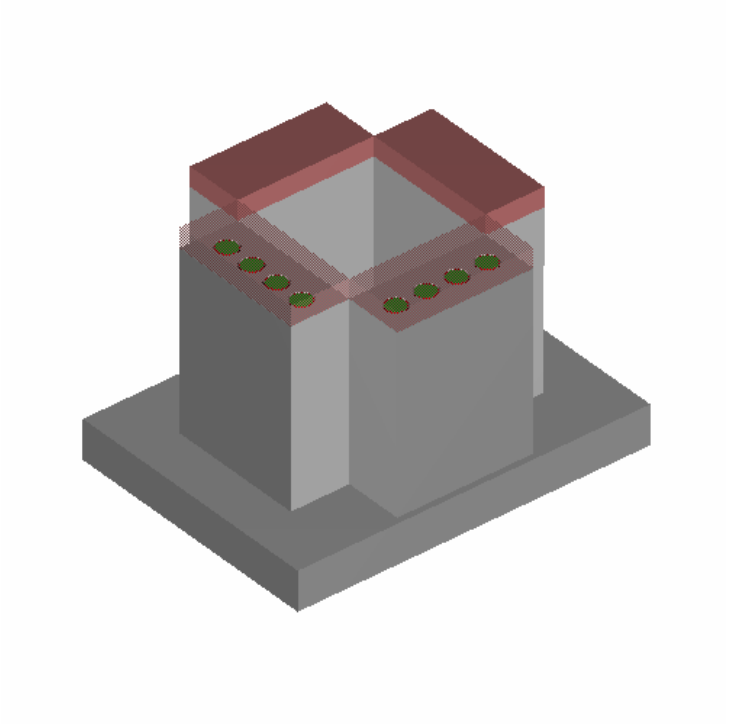


Figure 3. MCNPX model of PHNC standard configuration.

5. Results

The standard configuration, see Fig. 2, has an efficiency of 9% for ^{252}Cf in the center of the detector and a die-away time of 36 μs. The MCNPX model agrees with the measured data to better than 2%. The somewhat low efficiency is a result of the compromise between weight and efficiency, as well as the geometry picked for the standard configuration with throat diagonal of 25 cm compared to ^{3}He active length of 18 cm.

Figures 4 – 6 present the *x*-, *y*-, and *z*-axis profiles respectively {(*x*,0,0), (0,*y*,0), (0,0,*z*)}. Plotted in each graph are the measured Singles and Doubles rates, and the calculated Singles rate from the MCNPX model for comparison. In general the measured data compares well with the MCNPX model simulation, but there are discrepancies at the extremes of the model geometry. For example, in the case of the *x*-, and *y*-profiles the MCNPX model overestimates the count rate

for the case where the source is directly against the polyethylene moderator by about 5%. The overall trend of the data is as expected, namely an observed increase in efficiency as the source approaches the sides of the detector. For traditional coincidence or multiplicity counters designed for relatively fixed sample geometries, the position of the sample is restricted to the 'flat zone' – which in the case of the PHNC is only about 5 cm. However, the methodology for application of the PHNC does not require a uniform detection efficiency, but instead simply requires the ability to accurately model and thereby normalize to a standard configuration. In this respect the MCNPX model of the PHNC is acceptable for samples that all but touch the sides of the counter.

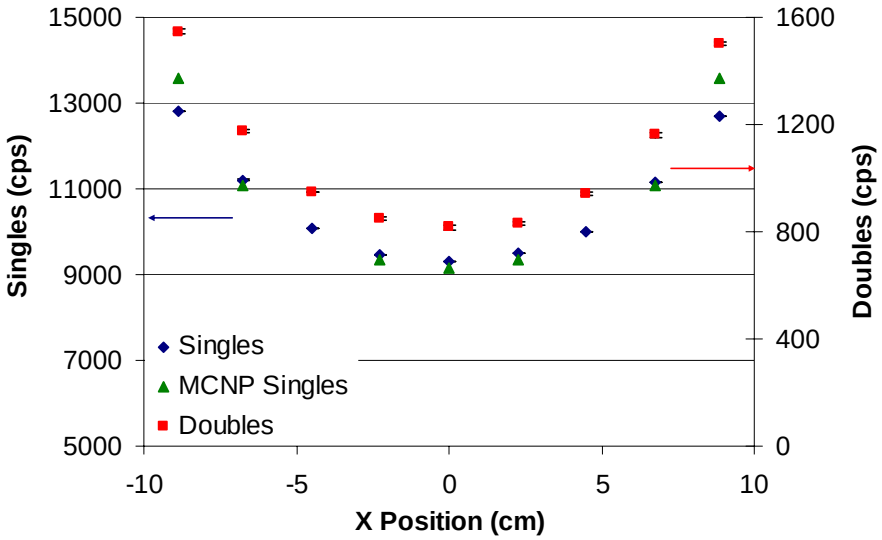


Figure 4. X-axis profile results.

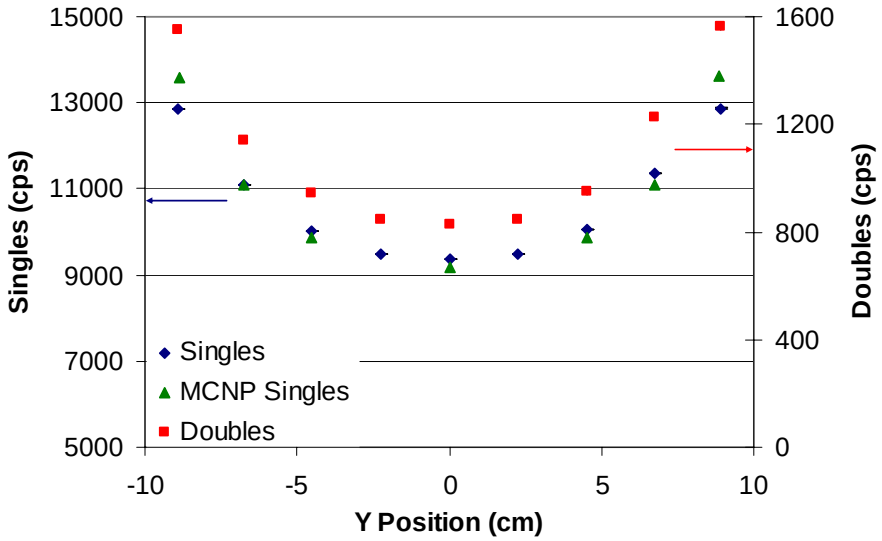


Figure 5. Y-axis profile results.

Figure 6 presents the *z*-axis response profile. The data suggest a large sensitivity to sample position along the *z*-axis and show the effect the bottom high-density polyethylene reflector as evidenced by the fact that the maximum efficiency is below the center of the ^{3}He tube active length by about 4 cm. In this case, the MCNPX model does not agree as well- in particular for the lower half of the profile. Part of this discrepancy is possibly the asymmetry of the ^{252}Cf source as a result of the encapsulation which is not included in the model. It is interesting to note though that the model performs quite poorly at the extreme position where the source is touching the bottom polyethylene reflector. In this position, non-isotropic distribution of the source would have a potentially large effect. We are planning to investigate this further.

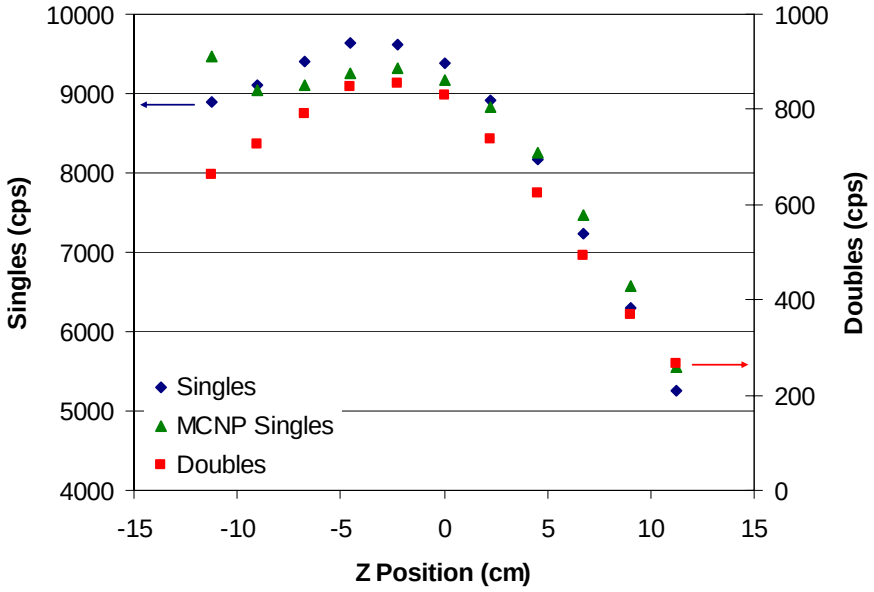
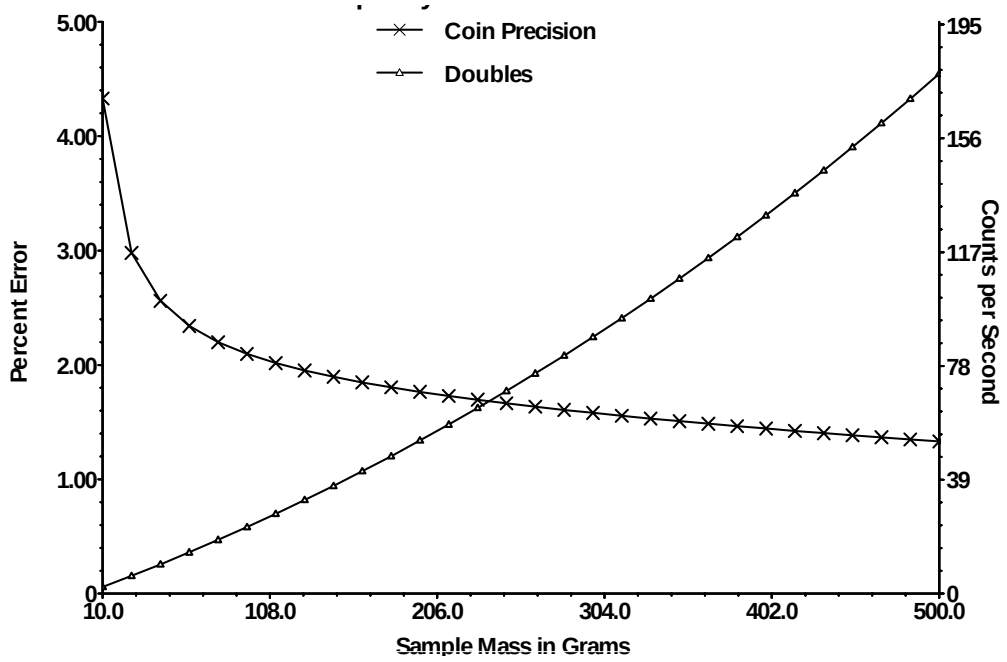


Figure 6. Z-axis profile results.

Although the model does not perform as well at the extreme areas of the geometry, namely at or very near the polyethylene moderator and reflector, agreement over the majority of the geometry is sufficient for most samples. This is because an actual sample will be distributed over a finite volume and not at a single point as in the case of the ^{252}Cf source. By virtue of being distributed, an averaging occurs in the response and reduces the effect of deviation of the model at a single point.

An estimation of the performance of the system in the standard configuration can be performed using the Figure-of-Merit technique of Ensslin [6], in which assay variance is employed as a measure of performance. Figure 7 shows results for the nominal case of a detection efficiency of 8%. As can be seen from Fig. 7, the PHNC is capable of attaining less than 2% precision on the Doubles count within a 300s count time for sample masses greater than about 10 g of plutonium. This level of performance makes the PHNC a valuable field-configurable assay system in addition to its use in Singles-mode surveys.



Once the detection system is characterized and calibrated for the standard configuration, measurements that are taken in any non-standard configuration can be normalized to the standard configuration using the MCNPX model.

6. Summary

A new portable handheld neutron counter, the PHNC, has been developed for field-configurable use in conditions where a flexible neutron detector is needed. The PHNC consists of a pair of ^{3}He detector slabs housed in briefcase and controlled by battery operated electronics and micro computer. Multiple detectors can be operated as a single detection system. The PHNC is applicable for large area surveys, source location and identification, as well as quantitative assay. A MCNPX model of the detectors in a standard configuration has been assembled and enables normalization of data taken in a non-standard configuration.

7. Future Work

Calibration of the PHNC in a variety of configurations, including the standard one, will be accomplished in the near term. This data will be compared to the results of the MCNPX model for benchmarking the field-configurable assay approach.

8. References

[1] L.S. Waters, Ed., “MCNPX User’s Manual, Version 2.4.0,” Los Alamos National Laboratory report LA-CP-02-408 (September 2002); J.S. Hendricks, G.W. McKinney, L.S. Waters, H.G.

Hughes, “New MCNPX Developments,” presented at International Meeting on Radiation Shielding and Protection, RPSD2002, American Nuclear Society, Santa Fe, NM, LA-UR-02-0642, April 14-17, 2002.

[2] Precision Data Technology, Inc., 601 40th Place, Everett, WA 98201.

[3] K. Bohnel, “Determination of Plutonium in Nuclear Fuels Using the Neutron Coincidence Method,” KFK2203, Karlsruhe (1975), also AWRE Translation 70 (54/4252), Aldermaston (1978).

[4] W. Harker, M. Krick, “INCC Software User’s Manual,” Los Alamos National Laboratory report LA-UR-01-6761 (November 2001).

[5] M.T. Swinhoe, J.S. Hendricks, and D.R. Mayo, “MCNPX for Neutron Multiplicity Detector Simulation,” 27th ESARDA Symposium on Safeguards and Nuclear Material Management (these proceedings, 2005).

[6] N.Ensslin, N. Dytlewski, and M.S. Krick, “Assay Variance as a Figure of Merit for Neutron Multiplicity Counters,” *Nucl. Inst. Meth*. **A290**, 197-207 (1990).

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The Absolute Calibration of Active Neutron Assay Instruments

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Abstract

Characterisation measurements of active neutron waste assay systems, such as the spatial mapping of the response within different matrices, can be performed using arbitrary fissile samples of convenient form factor. However, placing such measurements on an absolute scale requires cross calibration of these working samples to well defined reference items measured under similar conditions. Conventionally the centre of the empty drum is taken as the reference geometry. Dilute fissile reference standards for absolute calibration that are free from the effects of self-shielding and which are readily available are an idealisation but do not exist in reality. Real calibration samples must be corrected for self-shielding of the interrogating neutron flux. Self-shielding is a source of under-reporting and would result in a biased calibration if not allowed for. If the construction of a calibration sample is well known, the correction factors can be calculated. Alternatively if a range of samples are available then an experimental estimate of the self-shielding can be made.

In this work we describe the absolute calibration of differential die-away assay systems using a set of U_3O_8 reference materials (NBL CRM 969) originally designed for use as isotopic standards for use with gamma-ray spectrometry systems. The set comprises five samples each containing 200.1g of U_3O_8. They span the range from depleted to 4.5wt% enrichment. Taken as a series the samples allow the response under dilute conditions to be extrapolated directly. However, the self-shielding factors were also calculated using the Monte Carlo code MCNP™. Agreement, judged by the constancy of the count rate per effective fissile content, was excellent. The self-shielding factors were also compared with an algebraic formula developed previously and found to be useful for general estimates. The results were again favourable.

Additional calculations were performed for a second set of standards, NBL CRM 146. Three samples of 230g U_3O_8 were considered spanning the enrichment range of 20 to 93 wt%. Interest in these samples lies in their greater fissile mass content, which is needed for assay systems of poorer sensitivity.

We conclude that, from a characterisation perspective, the NBL CRM 969 and NBL CRM 146 sets of U_3O_8 are eminently suitable standards for the calibration of active systems even though they were conceived primarily as isotopic sources for gamma-ray spectroscopy systems. They are commercially in standard form and several Laboratories already have access to such sets. The sources are well described and meet the other essential requirements of samples suitable for absolute calibration.

1. Introduction

Assessing the inventory of fissile material in containerised waste is an important aspect of international Safeguards. Active neutron interrogation methods such as the Differential Die-Away (DDA) method [1-6] are sometimes used for this purpose. The DDA technique is a non-destructive approach for bulk analysis offering a favourable combination of high sensitivity and rapid throughput. The method is capable of high accuracy provided the conditions underpinning the calibration hold. Suitable calibration materials which are truly representative of the items to be measured are rarely available. Furthermore, special nuclear materials are extremely difficult to obtain and transport and this limits the scope of the calibration activities. Typically, therefore, the calibration rests on the assertion that the fissile material is present in dilute form so that self-shielding effects are negligible. This is a recognised reference condition. Allowance for deviation from this condition is usually made

during the reporting stage using independent information. The importance of self-shielding is well known and has been discussed in detail elsewhere [7-9] along with various methods that may be used to calculate the effect. Corrections for self-shielding often present a severe problem.

A procedure is described for obtaining an absolute calibration free from the effects of self-shielding of the interrogating neutrons in the sample holder and fissile material. The basic idea is to measure a series of nominally identical powders of U_3O_8 differing in the $^{235}U/U$ ratio so that the response per unit mass can be extrapolated to dilute conditions. The measurements are undertaken at the centre of the empty assay chamber and provide a datum against which to peg spatial profiles in surrogate matrices using a convenient arbitrary specimen which may have a more convenient form factor – for example small pellets or rods can be easily and quickly placed down re-entrant tubes in surrogate matrices with minimal perturbation on the response. The method uses a commercially available set of standards. Uranium is used as a surrogate for plutonium with basic nuclear data being used to provide the necessary link [10]. That is, results may be presented in units of either ^{235}U equivalent mass or ^{239}Pu equivalent mass. The residual effect of the sample holder is calculated by numerical methods in this work although in principle a test specimen measured in and out of the blank holder can provide the necessary ratio experimentally. This aspect will be confirmed by future work.

In subsequent sections we describe the experimental measurements, present the experimental findings and compare the result to calculations which make use of the knowledge of the reference materials used.

2. Assay Systems

In the DDA method, fast neutrons from a pulsed source are thermalised in the assay cavity. This interrogation flux persists far longer than the initial burst of fast neutrons and is therefore able to induce fast fission neutrons that can be detected in gated neutron detectors shielded from the thermal field. The measurements reported in this work were performed on five separate DDA systems. Four of the systems were of the Integrated Waste Assay System (IWAS) type and yielded essentially identical results. For the present purpose of expounding the general method we therefore use averaged values typical of a single determination. The IWAS instruments [11-13] integrate in a single assay chamber high resolution gamma spectrometry (HRGS) with passive neutron and active neutron counting capability. The gamma ray measurements, performed by two electrically cooled Canberra BE2820 Broad Energy Germanium detectors that are protected during the active neutron cycle to prevent neutron damage, are used to provide relative isotopic information via the Multi-Group Analysis (MGA) code and also to generate quantitative assay data which is complementary to the neutron techniques. The assay chamber is constructed primarily from high-density polyethylene (HDPE) moderator. The passive neutron detection efficiency is approximately 27%. The active neutron detection efficiency, resulting from cadmium wrapped ‘fast neutron detector packages’ embedded in the walls, is about 2.8%. The interrogation neutron field is provided by a 14MeV MF Physics Zetatron tube bursting at 100Hz with a time averaged yield of about $1x10^8$ $n.s^{-1}$. The generator is positioned in the corner of the cavity in a polyethylene reflector. Figure 1 shows two of the IWAS units undergoing factory characterisation and calibration.

The fifth instrument used was an integrated Passive Active Neutron Waste Assay System (PANWAS). Although of similar functional concept the design the construction of the cavity was entirely different. That is to say, although very similar nucleonics and software were employed the neutron physics portions were quite distinct. For this system the moderator material used was high purity graphite. The neutron detection efficiency was provided solely by cadmium wrapped HDPE moderated fast neutron detector packages [for the idea behind FNDPs see reference 14] so that a figure of approximately 26% was obtained in both passive and active modes for the matrix free drum. The Zetatron was housed in the rear wall in a lead booster and moderator assembly.

Figure 1 Two IWAS units undergoing calibration. The HRGS detectors are out of view on the left hand side of each chamber.

3. Reference Materials

In this paper we concern ourselves with two sets of certified reference material. The first set, NBL-CRM-969 formerly known and referred to here as NBS-SRM-969, comprises five samples plus an empty unsealed capsule and was available for measurement. The second set, NBL-CRM-146, comprises three samples together with a blank can for which we present calculations only.

National Bureau of Standards Standard Reference Material No. 969, NBS-SRM-969, is equivalent to EC NRM 171, European Community Certified Nuclear Reference Material No. 171. They provide well-defined bulk quantities of certified reference materials (CRM) in a well defined geometry. Table 1 gives a brief summary of the five sealed cans of U_3O_8 in each set. This information is taken from the detailed certificates of analysis and fabrication [15, 16]. The nominal ^{235}U abundance, 0.31, 0.71, 1.94, 2.95 and 4.46 mass % respectively, is used in the designation of the samples (e.g. we may use the notation NBS-446 and CBNM-446 interchangeably for the 4.5% enriched sample). The cans are made from ASTM-6061-T6 aluminium and contain 200.1g of oxide. The outer can diameter is 80mm and the can height is 89mm. The base has a well specified and controlled thickness of 2.00mm and serves as a lightly attenuating window for emitted γ–radiation. Figure 2 shows set NBS-SRM-969.

The oxide mass in each sample is known to $\pm$0.036% 1σ relative standard deviation. The U_3O_8 weight fraction is taken as (0.9975$\pm$0.00125) here based on a specification of >0.995. This corresponds to an overall 1σ uncertainty in the ^{235}U content of $\pm$0.14%.

The U_3O_8 powder is held in place by an aluminium top plug equipped with ultrasonic seals that provide a unique “fingerprint” for Safeguards purposes. The internal diameter of the can is (70.00 +0.05/-0.00)mm but for the present calculations we have adopted a value of (70.0$\pm$0.1)mm in a crude attempt to allow for limited powder non-uniformity.

Fill height, defined by the degree of compression applied to the powder by the plunger, for all samples is (20.8$\pm$0.5)mm except for sample NBS-446 for which the height is (15.8$\pm$0.5)mm. Because sample 446 is more compacted than the others, which all have the same shape, we calculated the self-shielding factors (SSFs) for both fill heights in this case – the true fill height and the fill height matching the rest of the set. The reason for this will become clearer in the results section in which the experimental results are plotted as a single set of identical shape (i.e. the only variable is self-shielding). This makes extrapolation to non-attenuating conditions (including the full set of samples) quite straight forward and effectively ensures NBS-446 is not an outlier (it turns out this does not have a dramatic effect on the ability to accurately extrapolate the curve, a shift of about 5% for the one data point in question, but we have taken this step as good practice).

**Figure 2 Photograph of the NBS-SRM-969 standards
The empty can is shown disassembled and the plunger in the foreground is missing the ultrasonic seal. The can laid on its side (second from the left in the back row) illustrates the thinned window. The steel rule at the side of the display is 150mm long.**

For some active neutron systems, such as Am/Li driven Active Well Coincidence Counters, the sensitivity may be far lower than for DDA systems like those employed here. In this case one may wish to have access to samples of higher ^{235}U mass to obtain a viable signal. To cover this eventuality we were led to consider also a set of NBL CRM-146 standards. There are three enrichments in the set plus an unsealed empty container. Details of the certification and fabrication can be found elsewhere [17, 18]. The nature of the encapsulation is rather similar to the NBL CRM-969 set. Each item of the New Brunswick Laboratory Certified Reference Material set, NBL CRM 146, contains 230.0g of U_3O_8 (controlled to about 0.1g and known to 0.04g to 0.12g typical) to a fill height of 15.8mm. The fill height is not a certified quantity. By comparison with the NBL CRM 969 we adopt here a fill height of (15.8±0.5)mm with 0.5mm again being the extreme variation. The ^{235}U mass loading for these three samples, taken from individual fill data records, is also summarised in Table 1.

Sample ID	Enrich-ment (wt.%)	Fill Height (mm)	U_3O_8 mass (g)	^{235}U mass (g)	RSD (%)
NBS-031	0.317	20.8	200.10	0.536	0.140
NBS-071	0.712	20.8	200.10	1.205	0.140
NBS-194	1.942	20.8	200.10	3.287	0.140
NBS-295	2.949	20.8	200.10	4.992	0.140
NBS-446	4.462	15.8	200.10	7.552	0.140
NBL-0017	4.462	20.8	200.10	7.552	0.140
NBL-0018	20.107	15.8	230.00	39.102	0.052
NBL-0019	52.488	15.8	230.04	101.770	0.049

Table 1 Description of the reference samples considered.

The mass of aluminium alloy in the unsealed can is about 748g. In the analytical model calculations described below we treat the can as a cylindrical jacket with a single thickness for the wall, base and top. For an oxide fill height of 15.8mm this corresponds to a thickness of 14.27(7)mm and for a fill height of 20.8mm to a thickness of 13.64(6)mm where we have taken the density to be 2.7g.cm^{-3}.

4. Calculations

Self-shielding factors were calculated by two techniques that have been described in detail elsewhere [9]. The first method employs an analytical approximation based on ENDFB-V cross sections whereby the SSF is expressed as an opacity weighted sum of the lightly attenuating and strongly attenuating idealised forms. This approach takes into account flux hardening (which sees the softer spectral components of the interrogating neutrons being preferentially removed by the outer zones) but assumes the incident spectrum can be represented by an ideal isotropic Maxwell-Boltzmann distribution evaluated at room temperature. The target material is treated as being purely absorbing i.e. scattering interactions in the sample are neglected. The effects of attenuation in the encapsulation are assumed to be small so that they may be treated independently by the application of a multiplicative factor estimated by a scaling rule of the type derived for the walls of ^{3}He-filled proportional counters [19]. These have been formulated by measuring the perturbation in the count rate of a compact spherical ^{3}He proportional counter (type Centronic SP9, 32mm internal diameter, 0.5mm stainless steel wall, filled with a partial pressure of approximately 250kPa ^{3}He), irradiated in a thermal neutron field, to additions of aluminium or 304 stainless steel shells.

The second method of estimating SSFs made use of the Monte Carlo N Particle, MCNP™, general purpose neutron transport code [20] with ENDFB-VI cross sections, according to a multi-step step process. The first stage involved calculating the energy spectrum of the interrogating neutrons by using the model to launch 14MeV neutrons from the Zetatron D-T generator and tracking then in the materials of the assay system. This spectrum from the PANWAS model and a Maxwellian flux distribution were used as the source term for subsequent calculations of the fission rate per incident particle. For the fission calculations, the encapsulated special nuclear material was modelled in detail according to the best physical description of the make-up and fabrication available. Additional calculations were performed with the same model but with the encapsulation and special nuclear material voided out. The flux tally from this run folded with the fission cross-section gave the ideal fission rate in the absence of self-shielding. The ratio of the two runs yielded the SSF for the sample. Runs with only the encapsulation void were used to estimate its impact alone.

In principle the Monte Carlo (MC) method is a powerful one in that the geometrical model can be fully detailed and an application specific interrogation spectrum can be calculated and used. Multiple scattering and multiplication are accounted for. Further more (surrogate) waste matrix and positional dependent (i.e. the dependence of where the source is placed in the calibration drum) effects can be evaluated. This would include flux perturbation effects caused by the presence of the sample. In practice extensive specific calculations of this kind are extremely time consuming and this is why we have adopted the multi-step approach outlined above and performed calculations for the sources in free space. For the present purposes this is not a limitation for waste applications because our objective is to establish an absolute response at the centre of an empty drum. This situation approximates closely to a uniform sea of probing neutrons. Other test sources can then be used to empirically investigate spatial and matrix effects directly as part of the systematic experimental study of volume weighted average response with matrix which underpins the formal characterisation of experimental matrix compensation function in terms of flux probes and matrix monitors. Being a numerical analogue simulation of the experiment, the MC technique is subject to statistical uncertainties. The cross-section data is also embedded in the method. This makes an analytical model more convenient to use for uncertainty sensitivity calculations, especially the propagation of the uncertainty on the SSFs due to the uncertainty in the macroscopic absorption cross section of the various materials.

5. Results

Each of the five samples was measured in turn at the centre of an empty 208 litre drum and the net count rate per unit fissile mass extracted. The absolute count rate depends on the output of the neutron generator (monitored across each run by flux and matrix probes) and on the neutronic characteristics of the particular assay system. For the present discussion these details are unimportant to the presentation of the concept and we therefore chose to normalise the response to unity for NBS-031 for all systems.

Figure 3 shows the neutron flux distribution from the PANWAS model used as source term of one of the sets of MCNP calculations of the SSF. The strength of the interrogating flux is concentrated in the range 10 to 100meV with a broad peak around 35meV.

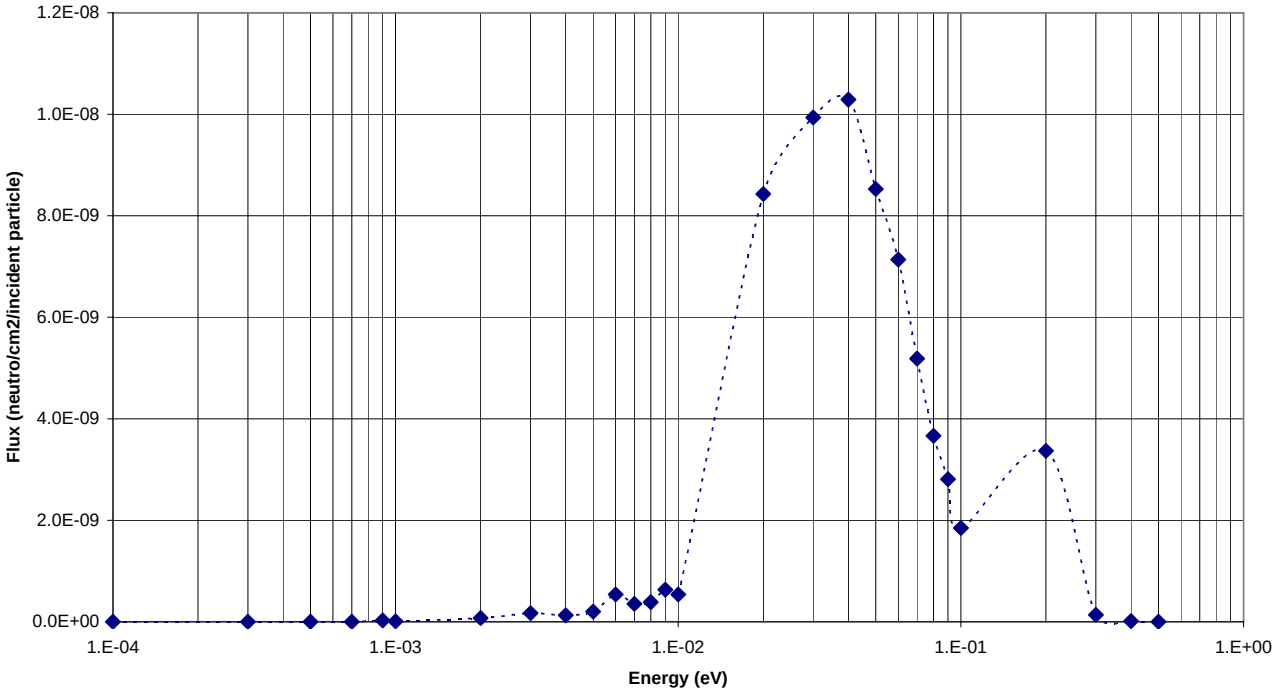

Figure 3 Plot of the calculated interrogating neutron spectrum in the graphite moderated assembly. N(E).dE is the fraction of neutrons in the energy interval dE about E.

The uncertainty analysis presented for the calculated SSFs was performed by varying the input parameters to the models within the ranges discussed and summing the deviations in quadrature. The MC and analytical methods yielded comparable fractional uncertainties for all of the key geometrical parameters such as fill height and radius with the oxide mass fixed. To assess the uncertainty associated with the absorption cross-section, taken as ±1% relative standard deviation the analytical model was used for convenience. It has been assumed that the uranium oxide is uniform – that is to say any point to point variations in packing density have been ignored. For the lightly attenuating samples, for which the interrogating neutrons have a mean free path that is comparable to the characteristic dimensions of the cylinder this is not expected to have a significant effect. For the most attenuating samples only the skin effect matters and again one can anticipate a minor contribution to the uncertainty. For intermediate cases the situation is difficult to judge but one might expect the method of production to result in a fairly consistent product.

The uncertainty on the experimental relative normalised specific response values includes allowance for counting precision on the signal counts, counting statistics on the flux monitors, uncertainty in the active background subtracted and also the uncertainty in the fissile mass content. Reproducibility uncertainties are small in comparison because the samples are placed close to the centre of the empty cavity where the spatial response is rather uniform.

Figure 4 shows the relative response from both monitors on a common scale and Figure 5 shows the relative DDA response predicted by the algebraic and numerical models. Numerical results may be found in Table 2.

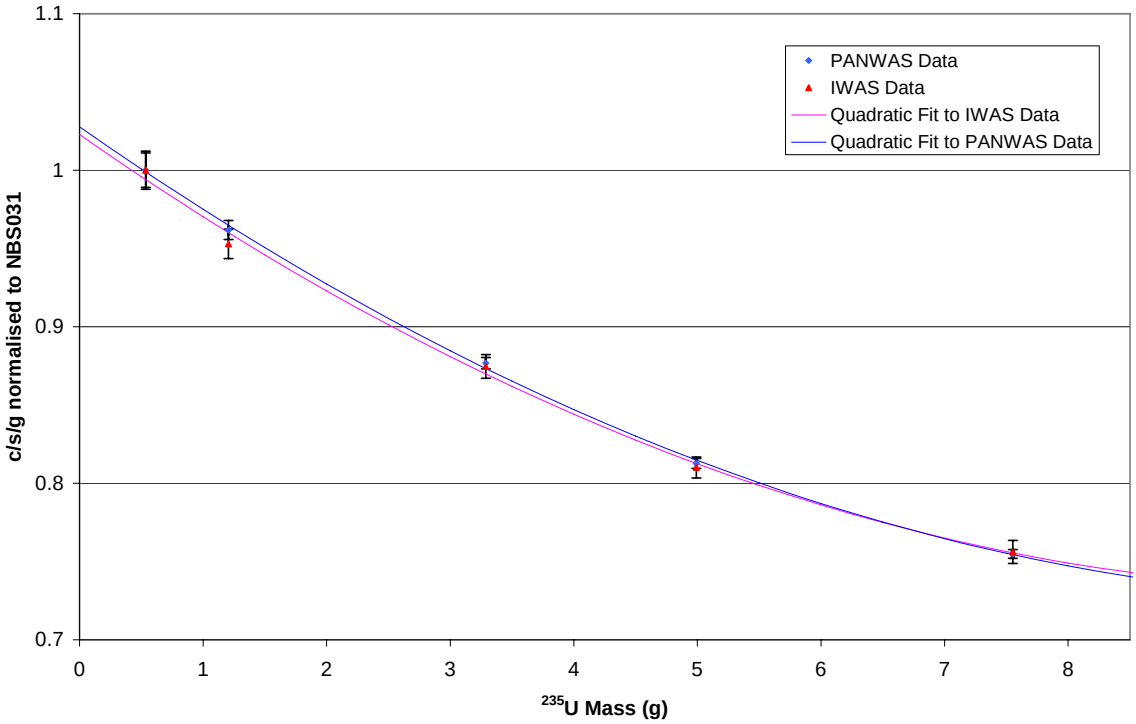

Figure 4 Comparison of the relative DDA response from each of the CBNM standards as measured on the two monitors

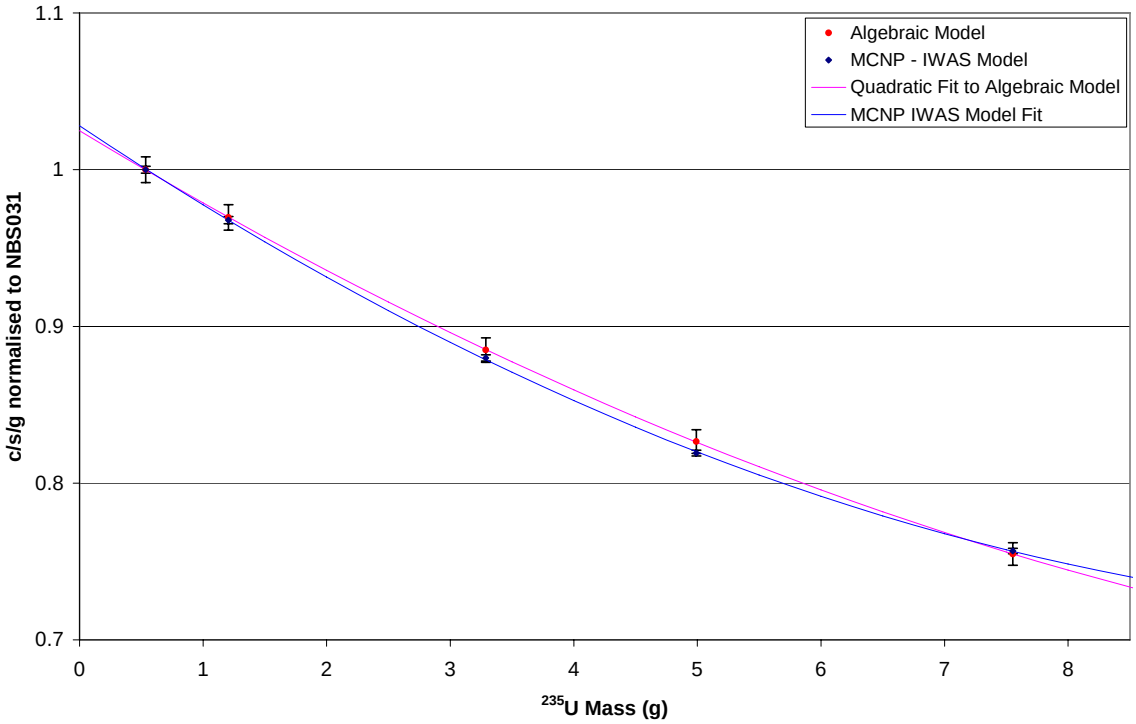

Figure 5 Comparison of the relative DDA response predicted using the algebraic and one of the MCNP models.

Sample ID	^{235}U mass (g)	RSD (%)	IWAS Measured Response ($c.s^{-1}g^{-1}$)		PANWAS Measured Response ($c.s^{-1}g^{-1}$)		IWAS MCNP Model ($c.s^{-1}g^{-1}$)		Algebraic Model ($c.s^{-1}g^{-1}$)	
			Norm.	Uncert.	Norm.	Uncert.	Norm.	Uncert.	Norm.	Uncert.
NBS-031	0.5359	0.1396	1.0000	0.0121	1.0000	0.0110	1.0000	0.0023	1.0000	0.0083
NBS-071	1.2050	0.1396	0.9530	0.0094	0.9617	0.0061	0.9678	0.0023	0.9695	0.0081
NBS-194	3.2870	0.1396	0.8747	0.0076	0.8767	0.0037	0.8798	0.0020	0.8849	0.0078
NBS-295	4.9916	0.1396	0.8101	0.0067	0.8127	0.0032	0.8192	0.0019	0.8265	0.0076
NBS-446[1]	7.5524	0.1396	0.7561	0.0074	0.7548	0.0028	0.7567	0.0018	0.7548	0.0072
Sample ID	^{235}U mass (g)	RSD (%)	-	-	-	-	-	-	Algebraic Model ($c.s^{-1}g^{-1}$)	
									SSF	Uncert.
NBL-0017	39.102	0.052	-	-	-	-	-	-	0.3480	0.0044
NBL-0018	101.770	0.049	-	-	-	-	-	-	0.1668	0.0026
NBL-0019	181.164	0.051	-	-	-	-	-	-	0.0971	0.0016

Table 2 Showing Data for graphs also given are the calculated results for samples NBL-0017, NBL-0018 and NBL-0019

Also shown in Table 2 are the calculated SSFs, inclusive of the wall effect, for samples NBL-0017, NBL-0018 and NBL-0019. Whereas the uncertainty in the calculated SSFs is less than ±1% for the low enriched set, the uncertainty for this higher enriched set increases to between 1.3-1.6% with the nuclear properties being more important than the treatment of the wall (in the case of the analytical model at least). Nevertheless, the uncertainties, even if they were to be applied directly in the calibration without extrapolation, are acceptable for waste assay applications. NBS-446 has an apparent mass of about 5g ^{235}U while NBL-0017 has an apparent mass of about 14g, a useful gain for systems of lower sensitivity. However, as the enrichment is increased beyond 20% the self-shielding becomes progressively more severe so that NBL-0018 and NBL-0019 both appear to contain 18g ^{235}U. In other words, there is an effective mass accuracy trade off that must be evaluated on a case by case basis but it may mean that enrichments greater than about 20% confer no significant advantage.

It can be seen that the IWAS and PANWAS measurement data sets agree very closely indicating that the SSF effect is predominantly the result of thermal neutron absorption in both cases. The curves, used to extrapolate to zero attenuation are the result of non-linear least squares fits to a second order polynomial with uncertainty allowances in both ordinate and abscissa. The general trend is reproduced very well across the full range. The curvature at low masses is quite modest which is consistent with the expectation of a linear behaviour in that regime. Therefore we have confidence that this empirical fitting procedure is adequate for extrapolating to determine the intercept value.

The plot of apparent specific response versus ^{235}U mass is equivalent to plotting the response against enrichment or attenuation length of the interrogating neutrons in the fissile material (because thermal neutron absorption in the ^{235}U dominates the macroscopic cross section). Extrapolation to zero ^{235}U mass loading therefore gives the limiting cnts/s/g calibration value for a non absorbing sample - that is one free of self-shielding in the fissile material. Because we have normalised the plots to unity for sample number 031 the value of the intercept is equivalent to the value of the self-shielding correction factor for that sample due to its nuclear material content.

All the data were measured with the sample container (can) around the uranium oxide and so the measured points do not account for the attenuation in the can. Therefore this has to be applied in addition. In the analytical model the attenuation factor due to the can (f_w) was calculated as:

$$f_w = a^t$$

where a is the attenuation per mm thickness and t the thickness (mm) of the can walls. The homogenised wall thickness was used as discussed earlier. Table 3 shows the SSFs calculated for the empty can wall. The effect, although not large, is significant. Unfortunately the uncertainty is sizeable owing to the difficulty in extracting a material specific scaling rule from the current set of

measurement data. As already noted, in principle this could be assessed experimentally using a separate compact specimen of fissile material placed inside the empty sample can.

Fill height	Algebraic Model		MCNP Model	
(mm)	SSF	σ(SSF)	SSF	σ(SSF)
15.8	0.971	0.008	0.957	0.001
20.8	0.972	0.008	0.969	0.001

Table 3 SSFs for the walls of the empty sample container – calculated using the algebraic model and MCNP (note that the uncertainty for the algebraic model is quoted at 1σ total, for MCNP only the 1σ statistical uncertainty is quoted)

Table 3 also shows the SSFs for the sample can walls calculated using MCNP. The magnitude of the can wall effect calculated by MCNP shows a small enrichment effect with the value of f_w increasing from 0.957±0.001 (for item 446) to 0.969±0.001 for the lowest enrichment sample (item 031). A mean value of 0.964 ±0.006 therefore covers the range. This value is also seen to be in good accord with the empirical scaling rule.

For items 017, 018 and 019 the SSFs for the capsule were estimated at 0.945, 0.940 and 0.940, respectively using MCNP. Any enrichment effect is not readily apparent.

Intercept values are summarised in Table 4. Along with the results obtained from the direct experimental measurements are listed values calculated using the results from the algebraic model and one of the MCNP models. The results from all of the approaches are essentially in perfect agreement being consistent within their evaluated standard deviation. Note that, in this table, the intercept including the can wall effect is based on the mean of the wall attenuation calculated according to the two methods (0.968 ±0.008).

Data Used for Fit	Intercept	Intercept including can effect
IWAS Measurements	1.0227 ± 0.0095	1.0565 ± 0.0131
PANWAS Measurements	1.0277 ± 0.0071	1.0617 ± 0.0114
Algebraic Model	1.0249 ± 0.0004	1.0588 ± 0.0085
MCNP IWAS Model	1.0280 ± 0.0014	1.0620 ± 0.0086

Table 4 Comparison of the intercept for the polynomial curves fitted to the measured and calculated data

6. Conclusions

Placing the fissile mass calibration of active neutron interrogation systems onto an absolute scale is a prerequisite for their application in safeguards and related technological applications. This can be difficult to achieve when representative reference materials are impractical to obtain and transport. In this work we have shown how relatively benign materials, in a configuration which is commercially available for other reasons, may be used simply yet with excellent results. The approach can generate a normalisation point for the specific response parameter for conditions of dilute fissile material based solely on experimental data. Five cans of U_3O_8 powder in a well-defined geometry and varying primarily in enrichment from depleted to lightly enriched (0.3 to 4.5wt%) allow an accurate extrapolation to be made. The samples of low enrichment are only lightly attenuating to the interrogating (thermal) flux and yet also conveniently provide an easily measurable signal. Transfer to units of $^{235}U_{equivalent}$ to $^{239}Pu_{equivalent}$ may be achieved using evaluated nuclear data. We have confidence in this step because, although not discussed in this paper, we have also gathered data using plutonium samples albeit with less well defined geometries and masses (the samples were of the PIDIE and CRM-136/137/139 type, details of which can be found elsewhere [21, 22]). As reported here we have additionally demonstrated the adequacy and agreement of two approaches to calculating self-shielding factors for the low enriched uranium oxide samples involved. For completeness we have also calculated self-shielding factors for a second set of reference samples not yet investigated experimentally using both the analytical approximation and the Monte Carlo simulation technique. The impact of the encapsulation has been calculated in this work but we recommend that this be assessed experimentally in the future by placing a ^{3}He proportional counter in a mock-up can or by assaying a small sealed fissile sample both in and out of the empty unsealed capsule provided with the reference set.

7. References

[1] P M J Chard and S Croft. Preliminary investigation of active neutron coincidence counting in Differential Die-Away assay. Proceedings of the 7th International Conference on Radioactive Waste Management and Environmental Remediation, Nagoya, Japan, 26-30 September 1999, ICEM '99 CD ROM ISBN 0-7918-1966-3, Session 32, Paper 3

[2] P M J Chard, S Croft and P B Sharp. Accurate matrix compensation for fissile material assay in drummed waste using a high sensitivity differential die-away chamber. 7th Nondestructive Assay Waste Characterization Conference, 23-25 May 2000, Salt Lake City, Utah Idaho National Engineering and Environmental Laboratory Report INEEL/EXT-2000-0002, Vol 2 (2000) 597-607.

[3] S Croft and L C-A Bourva. An Analysis of Gating Gains and Losses in Differential Die-Away Chamber Response Calculations. 8th Environmental Management Nondestructive Assay Characterization Conference, 11-12 December 2001, Denver, Colorado. US Department of Energy, Idaho Operations Office, Idaho National Engineering and Environmental Laboratory Transuranic and Mixed Waste Focus Area, Institute of Nuclear Materials Management. Report INEEL/CON-01-01581 (2002) 105-108.

[4] M Villani, L C-A Bourva, S Croft and R McElroy. The optimal relationship between differential die away counting gate settings and minimal detectable activities of fissile materials. Proceedings of 44th Annual Meeting of the INMM (Institute of Nuclear Materials Management), July 13-17 2003, Phoenix, Arizona, USA. Session E Paper 194. CD-ROM © 2003.

[5] M Villani, L C-A Bourva, S Croft and B McElroy. Optimal Differential Die-Away Early Gate Study for Difficult Assays. Proceedings WM'04, February 29-March 4, 2004 Tucson, Arizona, USA. Waste Management, Energy Security and a Clean Environment. HLW, TRU, LL/ILW, Mixed Hazardous Waste and Environmental Management. © WM Symposia, Inc. Paper WM-4248.

[6] L C-A Bourva, S Croft, B McElroy and M Villani. Evaluation of the minimum detectable quantities of fissile material in a differential die-away chamber. Waste Management 2003 Symposium, WM'03 Conference, February 23-27 2003, Tucson, Arizona. CD ROM © 2003 Symposia, Inc. PO Box 35340, Tucson, AZ 85740. Session 18D.

[7] P M J Chard and S Croft. Self-sheilding factors for the Cf-shuffler neutron interrogation technique. 17th Annual ESARDA (European Safeguards Research and Development Association) Symposium on Safeguards and Nuclear Material Management, Aachen, Germany, May 9-11, 1995. ESARDA 27 EUR 16290 EN(1995) 557-562.

[8] P M J Chard, L C-A Bourva and S Croft. A Review of Self Shielding Effects in Active Neutron Interrogation 23rd Annual Meeting ESARDA (European Safeguards Research and Development Association) Symposium on Safeguards and Nuclear Material Management, Bruges, Belgium, 8-10 May 2001. EUR 19944 EN (2001) 540-551. ISBN 92-894-1818-4.

[9] L C-A Bourva, S Croft and P M J Chard. The Effect of Self-Shielding in Active Neutron Assay of Plutonium-Bearing Aggregates, 8th Environmental Management Nondestructive Assay Characterization Conference, 11-12 December 2001, Denver, Colorado. US Department of Energy, Idaho Operations Office, Idaho National Engineering and Environmental Laboratory Transuranic and Mixed Waste Focus Area, Institute of Nuclear Materials Management. Report INEEL/CON-01-01581 (2002) 116-122.

[10] P M J Chard and S Croft. A database of $^{240}Pu_{effective}$ and $^{235}U_{effective}$ coefficients for various fertile and fissile isotopes. 19th Annual ESARDA (European Safeguards Research and Development Association) Symposium on Safeguards and Nuclear Material Management, Montpellier, France, May 13-15, 1997. ESARDA 28 EUR 17665 EN(1997) 389-396.

[11] R McElroy, S Croft, B Young and L C-A Bourva. Design and performance of the Integrated Waste Assay System (IWAS). Proceedings of 44th Annual Meeting of the INMM (Institute of Nuclear Materials Management), July 13-17 2003, Phoenix, Arizona, USA. Session F Paper 218. CD-ROM © 2003.

[12] R McElroy, S Croft, B Gillespie and B Young. Total measurement uncertainty analysis for an integrated waste assay system. Proceedings of 44th Annual Meeting of the INMM (Institute of Nuclear Materials Management), July 13-17 2003, Phoenix, Arizona, USA. Session H Paper 221. CD-ROM © 2003.

[13] R D McElroy and S Croft. Integrated Waste Assay System (IWAS) Analysis Enhancements. Presented at Waste Management, Tuscon, AZ, Feb.2005.

[14] S Croft, L C-A Bourva, P M J Chard and S Melton. Principles of Fast Neutron Detector Package Design for Differential Die-away Technique Assay Systems. Proceedings of the 43rd Annual Meeting of the Institute of Nuclear Materials Management, Orlando, Florida, June 23-27 2002. CD ROM © 2002 Documation, LLC. Session A: Tuesday Afternoon 25th June. Material Control & Accountancy –Neutron Measurements. Paper #371.

[15] National Bureau of Standards Certificate. Standard Reference Material 969: Uranium isotopic standard reference material for gamma spectrometry measurement. (In cooperation with the Commission of the European Communities, Central Bureau for Nuclear for Nuclear Measurements, Geel, Belgium, and the U.S. Department of Energy, New Brunswick Laboratory, Argonne, Illinois. NBS SRM 969 corresponds to EC CRM 171.). Certificate NBS-107, NBS, Gaithersburg, USA, MD. Revision dated 6-27-85.

[16] Commission of the European Communities, Joint Research Centre, Geel Establishment (CBNM), Certified Nuclear Reference Material Certificate of Analysis. EC Certified Nuclear Reference Material 171 ^{235}U Isotope Abundance Certified Reference material (U_3O_8) for Gamma-Spectrometry. (In cooperation with NBS, Gaithersburg, MD, USA. EC CRM 171 corresponds to NBS SRM 969). CBNM, Geel, Certificate of Analysis dated 19 June 1985

[17] New Brunswick Laboratory Certified Reference Material Certificate of Analysis CRM 146: Uranium isotopic standard for gamma spectrometry measurements. U.S. Department of Energy, NBL, Argonne, Illinois, USA (July 30, 1999).

[18] Mary Ellen Downey, Nuclear Reference Materials Specialist, U.S. Department of Energy, New Brunswick Laboratory. Priv. Comm. 1st Sept. 2004. CRM 146 NBL 0017, 0018, 0019 containment and assembly details. See also NBL-354 (March 2000).

[19] S Croft and J Chapman. Practical Considerations for Selecting Cylindrical ^{3}He-filled Proportional Detectors for Homeland Defense Applications. 45th Annual Meeting of the INMM. Paper 393(July, 2004).

[20] J F Briesmeister (Ed.). MCNP™ – A general Monte Carlo N-Particle transport code, version 4B. Los Alamos National Laboratory report LA-12655-M, March 1997.

[21] P M J Chard, S Croft and P B Sharp. Characterisation of the Harwell N95 high efficiency passive neutron counter. 17th Annual ESARDA (European Safeguards Research and Development Association) Symposium on Safeguards and Nuclear Material Management, Aachen, Germany, May 9-11, 1995. ESARDA 27 EUR 16290 EN(1995) 551-556.

[22] R Venkataraman and S Croft. Determination of Plutonium Mass Using Gamma Ray Spectrometry. Nuclear Instruments and Methods in Physics Research A505 (2003) 527-530.

Performance characterisation for NDA systems used for sentencing and verifying LLW containing fissile material, at Dounreay

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Abstract:

The UKAEA operates a Low Level Waste (LLW) receipt, assay, characterisation and supercompaction facility (WRACS) at its Dounreay site. This facility provides a site – wide service for the processing of LLW consigned from various facilities, culminating in supercompaction prior to storage.

The radionuclide and fissile inventory of 200 l LLW drums is strictly controlled at the point of generation, based on a robust quality system. WRACS then provides a quality check, to verify that the inventory is consistent with the declaration, and conforms to the plant's conditions for acceptance (both physical and fissile / radionuclide characteristics). A suite of instrumentation is used, comprising a Real Time Radiography (RTR) system, a Segmented Gamma Scanner (SGS) and a Passive / Active neutron interrogation system. The SGS is used to check the β/γ radionuclide content, while the neutron system checks the fissile ^{235}U and Pu inventory. In order to be accepted for processing in WRACS, the radionuclide and fissile inventory must be consistent, within fixed limits, with the declaration.

In establishing the optimum acceptance criteria, it is vital that the performance of the NDA instrumentation is fully understood, with respect to the measurement uncertainty. For example, if the levels of agreement demanded between the WRACS NDA results and the consignor's declarations, are too restrictive, then a substantial number of false rejects will occur due to natural statistical variations, leading to interruptions to normal plant operations.

UKAEA's NDA section has recently undertaken a comprehensive performance assessment for each NDA system in WRACS, based on current best practice methodologies. This includes an assessment of the detection limits and measurement uncertainties for ^{235}U assay, throughout the full dynamic range of operation, based on the results for a set of test drums and test sources / fissile standards. By comparison with a similar assessment for the NDA systems used to consign ^{235}U – bearing LLW from a specific plant, we have been able to optimise the consigning plant's drum loading fissile limits, thus greatly minimising the number of WRACS reject drums, improving the efficiency of the overall waste management operations.

In this paper, we describe the technical methodology behind these performance assessments, focussing on ^{235}U assay. We also compare the measurement uncertainties, with plant experience for a large number of real waste drums, comparing the ^{235}U results from the SGS and the neutron system. This study provides valuable confirmation of the validity of the performance evaluations, and a valuable insight into the characteristics of real wastes.

Keywords: 235U, uncertainties, waste, assay, NDA

1. Introduction

The United Kingdom Atomic Energy Authority (UKAEA) owns and operates a Low Level Waste (LLW) Receipt, Assay, Characterisation and Super-compaction (WRACS) facility. 200 *l* waste drums sentenced from a wide range of consigning facilities at Dounreay, are characterised by real time radiography and assayed by Non Destructive Assay (NDA) systems, prior to supercompaction and storage in half-height ISO containers. Full operations commenced during the summer of 2001.

The role of the NDA systems is to provide a robust quality check to confirm (or otherwise) the validity of the fissile material and radionuclide inventory information declared by the consignor, for each drum. These systems include a Real Time Radiography (RTR) set (to confirm that the physical composition of each drum is acceptable), a segmented gamma scanner (known locally as NDA19) and a passive/active neutron (PAN) interrogation system (NDA20),

The uncertainties associated with NDA measurements inevitably leads to discrepancies between the consignor's declaration, and the results obtained by the WRACS NDA systems. It is therefore vital that the performance of the NDA systems is well understood. Otherwise, there is no sound basis for judging the confidence that can be placed on a particular set of results.

WRACS is a fully operational plant with challenging throughput targets. If it is necessary to reject drums due to excessive discrepancies between the consignor's declarations and the WRACS NDA system results, a comprehensive investigation is launched. This leads to an improved understanding of the properties of the rejected consigned waste, which can be fed back into the waste characterisation process for the plants. It is clearly desirable to have a high degree of confidence that drums are rejected due to statistically significant discrepancies. In order to minimise the number of "false" rejects, it is therefore necessary to allow an appropriate tolerance, which properly allows for the uncertainties in the NDA systems both at WRACS and at the consigning plants. By adjusting this tolerance in line with the evaluated measurement accuracy of the various NDA systems, it is possible to optimise the throughout of the WRACS plant, whilst still maintaining adequate scope for detecting significant discrepancies.

In this paper, we describe a rigorous performance evaluation of both NDA systems deployed in WRACS, leading to robust detection limit and measurement accuracy results, based on a comprehensive set of commissioning measurements performed using surrogate waste matrix – filled drums, with test sources positioned at various locations inside each drum. This evaluation, carried out according to established UK industry best practice formalism, leads to a sound understanding of the performance characteristics of the NDA systems. We use the example of a specific Dounreay plant that consigns ^{235}U waste to WRACS, as a case study, describing how our robust performance evaluations have been used to tailor the acceptance criteria for the WRACS plant, and have lead to a substantial reduction in the number of "false rejects".

We also present an analysis of the WRACS NDA results for a large number of real waste drums that have been processed through WRACS since 2001. By comparing the ^{235}U mass results from NDA20 and NDA19, it is possible to assess the measurement accuracy under typical operational conditions, rather than ideal laboratory conditions. This assessment can provide confidence in the validity of the original performance assessment, and valuable insight into the variations for real wastes, to refine the performance assessments.

2. Mode of operation of NDA systems

A Segmented Gamma Scanner (NDA19) and Passive / Active Neutron Interrogator (NDA20) operate in WRACS. These are designed to provide quality checks to confirm the validity of the β/γ radionuclide activities and fissile material mass, respectively, declared by the consigning plant [1].

NDA19 (Figure 1) comprises 4 electrically cooled HpGe detectors, mounted in collimators to view the drum in 4 axial segments. The drum rotates continuously, to achieve a uniform spatial response profile. A ^{152}Eu transmission source is mounted diametrically opposite each detector, behind a computer – controlled shutter shield. These provide matrix compensation functionality, separately in each segment. The activities are reported for library – selected key nuclides, and the total activity is reported. The ^{235}U mass is also reported by the analysis software, although this is only used occasionally, when the primary result from NDA20, is called into question.

NDA20 (Figure 2) is a combined Passive / Active Neutron Interrogation system [2, 3]. Passive Neutron Coincidence Counting (PNCC) is used to measure the $^{240}Pu_{effective}$ mass. This is then converted to a total Pu specific alpha activity inventory, using pre-selected isotopic fingerprint data, chosen to adequately represent the operational conditions of the consigning plants. Acceptance levels are imposed, to ensure both that the measured total Pu alpha activity is below a fixed threshold, and that the measured and declared activities are in reasonable agreement. The system operates as a californium shuffler in active mode. This measures the total fissile mass, which comprises, principally, ^{235}U, and is therefore expressed as a ^{235}U effective mass. As for the total Pu alpha activity, there are acceptance thresholds on the measured ^{235}U mass and the discrepancy between the measured and declared values. Three fixed irradiation positions are employed, to ensure uniform irradiation of the drum, thereby minimising the assay uncertainty as a result of non-uniformity in the spatial distribution of active material. The drum is mounted on an indexed turntable, and a set of matrix monitors is deployed [2], to correct for the effect of the matrix on the passive and active mode response. Additionally, under certain conditions, corrections are made [2] for spatial non –uniformities in the distribution of Pu / U, through the application of a set of algorithms developed during the instrument design.

The NDA systems provide a quality checking function, in order to confirm (or otherwise) the validity of the consignors declaration. They are rarely used for providing direct sentencing information. NDA20 has always been used to provide this function for α activity (principally ^{235}U, but also checking the Pu content), recognising that neutron measurements are, in general, less susceptible to gross under-estimation in dense matrices, than gamma measurements.

3. Calibration approach

Waste is consigned from facilities on site including historical waste pits, post-operation clean-out of active areas, general low level waste from clean-up operations, decommissioning waste, and repackaged historical waste. The waste matrices range from low density, soft wastes, to dense metallic matrices. This diversity of waste forms, places considerable demands on the NDA systems.

The systems were calibrated and tested using a set of surrogate homogeneously filled waste matrix drums, containing re-entrant tubes into which radionuclide and fissile material test sources could be inserted at reproducible positions. This allowed the response of the systems to be determined for a wide range of matrices, and locations for the active material. These drums were carefully selected to represent the full range of the expected properties of real waste forms in WRACS. However, it is recognised that there are limitations in the validity of this approach, for example due to the expected in-homogeneities in real waste matrices.

In the case of NDA20, the analysis algorithms were developed based on a comprehensive set of test measurements using the surrogate waste drums. In the case of NDA19, the matrix attenuation correction in each segment was determined from empirical attenuation calculations, with reference to the case of an empty drum.

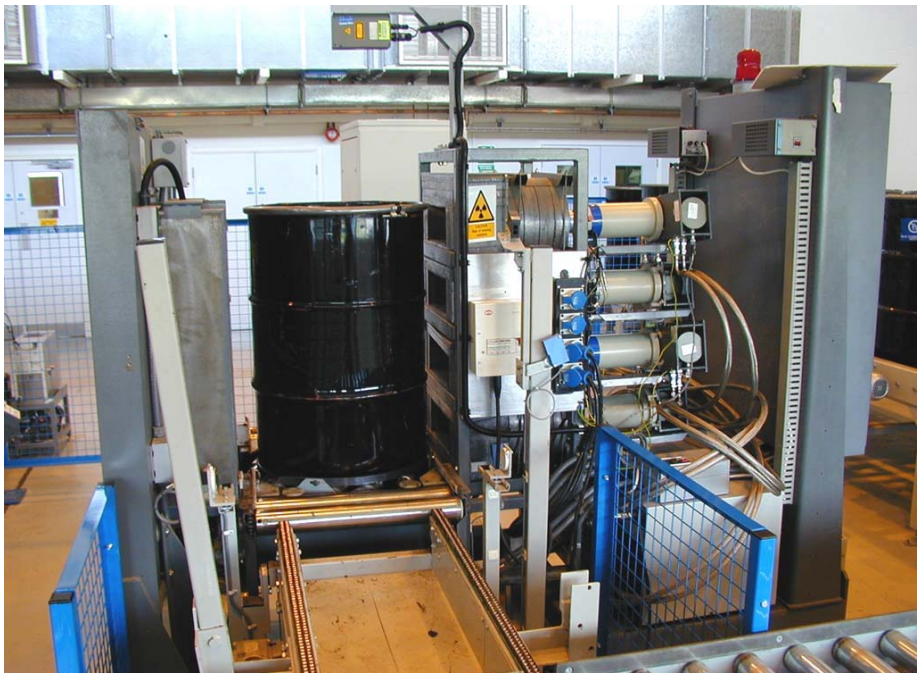

Figure 1. NDA19 Segmented Gamma Scanner

Figure 2. NDA20 Passive / Active Neutron Interrogator

4. Performance Evaluation Methodology

UKAEA / Dounreay operates and manages all the site's NDA systems according to a robust quality system based on established UK industry best practice principles [4]. This is designed to provide confidence for both the plant management and regulatory authorities, in the validity of the results produced by the NDA systems. An important element is the standardisation of practices for evaluating, and expressing, key measurement performance parameters. This removes any potential for ambiguity with regard to the exact meaning of quoted performance data.

Below we outline the general approach to performance evaluation, for the various NDA systems at Dounreay. We then describe how this approach has been implemented for NDA19 and NDA20, and also for a set of NDA systems deployed in a plant that consigns ^{235}U – contaminated waste to WRACS.

Both the detection limit and measurement accuracy are of interest, for various operational reasons. In order to determine the measurement accuracy, we appeal to the results of the comprehensive commissioning program for the NDA system in question. By performing a comprehensive analysis of the measurement performance, for test sources / fissile standards placed at fixed locations in the range of matrix – filled 200 ℓ drums / test packages, it is possible to determine the measurement accuracy under these conditions. Underpinning this analysis approach, is the assumption that the range of matrices studied during these tests, is representative of the full range of waste matrix properties, and radionuclide / fissile material spatial distribution patterns, to be expected in LLW real waste arisings.

The following contributory factors must be considered [4], in performing an assessment of the measurement accuracy:

- Counting statistics
- Effect on counting statistics, of typical variability in plant background
- Accuracy of activity / fissile mass value for calibration standards
- Effect of differences in the physical / chemical form of real waste material, compared to those properties for the calibration sources / standards (e.g. self-shielding)
- Effects of the waste matrix on the instrument response. Note that NDA19 and NDA20 perform matrix compensation measurements and algorithms, so this factor represents a combination of the magnitude of the matrix effect, and the performance of the compensation technique. For other systems, fixed calibration curves exist for discrete calibration matrices.
- Effects of non – uniform spatial distribution of activity within a waste drum / package.

4.1. NDA19

As part of the setting – to – work commissioning, a set of test measurements were performed using 4 test matrices, comprising a) empty drum gross density ≈ 0.1 $g.cm^{-3}$, b) soft waste, predominantly cellulose, gross density ≈ 0.4 $g.cm^{-3}$, c) graphite pieces contained in aluminium cans, gross density ≈ 0.4 $g.cm^{-3}$, d) concrete, gross density ≈ 1.0 $g.cm^{-3}$. The soft and graphite matrices comprised waste placed into, respectively, 195 aluminium cans, and 57 tin-plate cans. The concrete matrix comprises solid concrete blocks. The drums were specially prepared, each incorporating 3 vertical re-entrant tubes, at the drum centre, at half the drum radius from the drum centre, and at the edge of the drum, allowing measurements to be performed at a wide range of positions inside each drum. These simulated waste matrix drums are considered to represent an extreme range of expected properties of real waste drums, covering the full range of masses (up to ≈ 200 kg) with realistic inhomogeneity.

The commissioning measurements produced a dataset covering the full energy range for which NDA19 is being used. For the present purposes, we appeal to the ^{133}Ba results, which cover the approximate energy range, to include the principal 186 keV line from ^{235}U.

The SGS calibration assumes a uniform spatial distribution. We chose to study the effects of non-uniform spatial distributions of ^{235}U, by investigating the variability of response for a single source,

positioned anywhere within the drum. A weight was applied, when calculating the mean and standard deviation of the population of results for each drum, proportional to the volume of the annual region represented by each tube position. The results demonstrate that this represents the largest source of uncertainty.

The volume – weighted average (VWA) result, expressed as a fraction of the true source activity, gives a measure of the performance of the matrix compensation. In view of the small number of matrix drums, it was decided to quote this in terms of a bias value, for each matrix. By analysing the distribution of results over all the sample position measurements in all the tests drums, it is possible to derive an indicative uncertainty figure for an "unknown drum".

The 1σ uncertainty from position effects, ranges from $\approx$ 29 % for an empty drum, to 80 % for the concrete matrix drum, with a value of 46 % for a general unknown drum. The 1σ contributions from the calibration (2%) and counting statistics (10 %, expressed for the 1×10^5 Bq ^{133}Ba test source) represent a negligible proportion. The calibration uncertainty originates from the "goodness of fit" of the efficiency calibration curve, compared to the true measured efficiencies. Adding each component in quadrature, gives the overall total measurement uncertainty (TMU). By treating the bias value (from matrix effects) as a 1 σ component, this approach leads to a conservatively high overall TMU, but this is a small effect, as the TMU is dominated by position effects (see Figure 3).

The overall 2 σ TMU results are 71 %, 166 %, 89 %, 99 % and 74 % for the empty drum, concrete matrix drum, soft waste matrix drum, unknown matrix drum, and the graphite matrix drum, respectively.

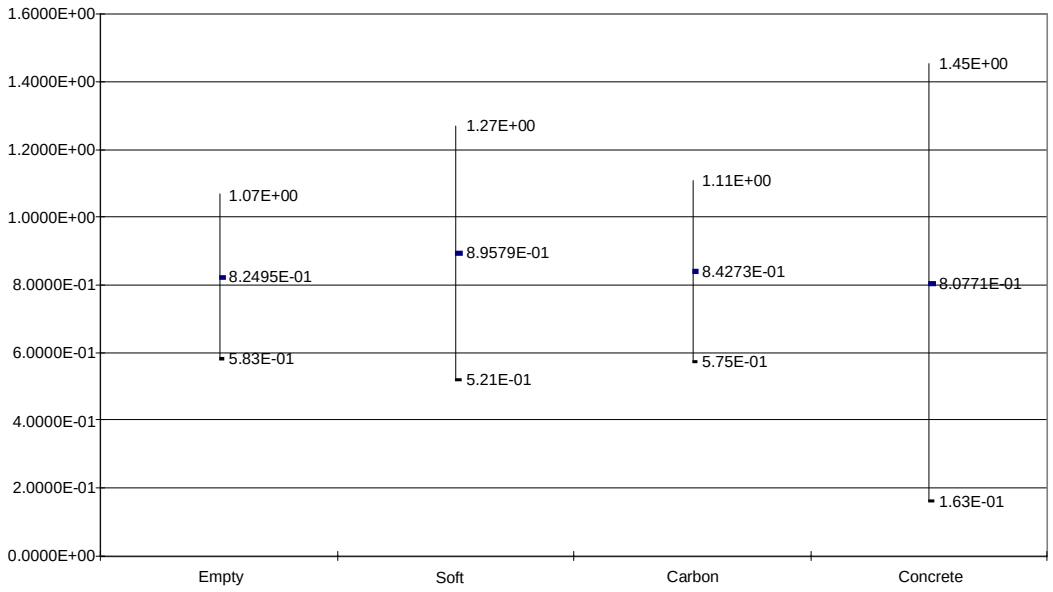

Figure 3. NDA19 uncertainty results for ^{133}Ba, indicative of ^{235}U. The measured / true activity is plotted as a function of matrix, the error bars representing positional variability (1σ).

4.2. NDA20

NDA20 [2] is based on a dual passive and active interrogation cycle, measuring the ^{240}Pu effective and ^{235}U effective masses respectively. A full uncertainty assessment has been performed for both measurement modes, but for the present discussion we shall restrict ourselves to the active mode.

As part of the system calibration program, measurements were performed using 11 simulated waste matrix drums. These were selected to cover the full range of expected waste matrices in WRACS, both in terms of the matrix composition, bulk density, and homogeneity.

The system incorporates a matrix compensation algorithm, based on the use of 3 pairs of External Matrix Monitors (EMM's), mounted on the side of the drum opposite the ^{252}Cf irradiation position. This provides matrix compensation in 3 axial segments. Each EMM pair measures the thermal (bare ^{3}He) and fast (Cd shielded) neutron flux, which is diagnostic of the perturbation of the drum matrix on the induced fission rate. The system also incorporates an axial position compensation algorithm. By comparing the shuffler results for the three separate ^{252}Cf irradiation heights, it is possible to apply a crude compensation for axial asymmetries in the spatial distribution of fissile material. A matrix inversion technique is employed in order to correctly accommodate the "cross – terms" arising from induced fissions in ^{235}U present in the other 2 axial positions, not matching a particular ^{252}Cf irradiation height.

NDA20 was calibrated using a different approach from that of NDA19, therefore a modified approach to the uncertainty assessment was required. The algorithms incorporate a set of look-up tables, in which specific diagnostic information is related to the calibration count rate parameter, i.e. the count rate per unit mass ^{235}U (on a fixed reference date). For example in the case of the matrix compensation, the EMM "Fast/Thermal" ratio is related to the c/s/g ^{235}U calibration parameter, for the 11 calibration drums. The entries for the 11 drums are sorted in increasing order of F/T, and linear interpolations are performed to determine the calibration parameter for an unknown real waste drum, based on the measured Fast/Thermal ratio. Ideally, the ratio would exhibit a linear fit to the c/s/g ^{235}U calibration parameter. However, in practice the EMM monitors response ratio does not perfectly follow the induced fission rate within the matrix, due to various factors (notably matrix inhomogeneities, and differences in energy distribution of the induced fission rate and the neutron spectrum impinging upon the EMM's). It is possible, for example, that two matrices may have similar EMM response ratios (Fast /Thermal ratio), but different calibration parameters (c/s/g ^{235}U). It was therefore decided to analyse the variability of the calibration parameters, about the ideal situation of a straight line "best fit". This gives an indication of the likely variability in the matrix compensated, calibration parameter.

The matrix compensation algorithm performance was analysed initially, for a uniform ^{235}U distribution within the matrix. The positional variability was then analysed, in a similar manner as for NDA19. The average uncertainty was calculated, for the 11 calibration matrices.

Since the majority of the ^{235}U contained within waste consigned to WRACS, takes the form of highly dispersed surface contamination, the calibrations were originally performed using a thin foil of enriched uranium metal. Monte Carlo modelling was performed using the MCNP^{TM} code [5], to correct for the inherent self-shielding in this standard, so that the calibration results apply, strictly, for dispersed fissile material.

1σ uncertainty components of 2.5, 2.5 and 7.1 % associated with the ^{235}U reference standard mass, the self shielding correction factor, and the ^{252}Cf source strength, respectively, were estimated. The 1σ uncertainties from matrix effects, was calculated to be 19.9 % (applicable for a uniform ^{235}U distribution). The 1σ contribution from axial and radial position effects is 18.8 %, decreasing to 17 % if the axial compensation algorithm is applied (the uncertainty is dominated by the radial position effect). The overall 1σ TMU (excluding counting statistics) is therefore 27.4 % and 26.2 % for cases without and with the axial compensation algorithm, respectively. The detection limit is $\approx$ 0.27 g ^{235}U, for a ^{252}Cf source $\approx$ halfway through its normal operating life. For typical gram quantities of ^{235}U consigned in 200 *l* drums to WRACS, the counting statistics contribute a negligible amount to the overall TMU.

The overall 1 σ TMU result is therefore $\approx$ 27 %, for a general unknown matrix, for typical gram quantities of ^{235}U, such that the counting statistics do not contribute a significant amount.

Further to the above uncertainty assessment, based on the assumption of dispersed ^{235}U, MCNP calculations were performed in order to predict the typical under-estimation expected due to self-shielding. Calculations were performed [6] according to the formalism described in [7], modelling typical swab samples of waste, containing up to 10 g ^{235}U. For typical swab composition and dimensions, the results indicated that an under-estimation of approx 10 – 20 % could be expected.

This represents the most extreme bias expected from this effect, which would occur if ≈ 10 g ^{235}U arose in a single swab sample. It was not considered necessary to apply a further correction for this effect, since it is smaller than the matrix and positional uncertainties.

4.3. NDA systems in a typical consigning plant

The uranium recovery plant at Dounreay, produces waste arisings from both routine operations, and various decommissioning operations. Due to the diversity of waste streams encountered throughout the various processes (swabs / calcium fluoride slag, filters, graphite crucibles, metallic waste, etc), a suite of NDA systems is deployed, implemented in a complementary fashion, for measuring the ^{235}U content and enrichment, of individual waste packages. Known plant fingerprints are used in conjunction with the measured ^{235}U mass, to provide robust activity assessments which are used to sentence the waste to the appropriate storage / treatment facility at Dounreay. For typical soft waste, which is not expected to contain significant quantities of ^{235}U, a Low Resolution Gamma Spectrometer "Package Counter" is used to measure the ^{235}U inventory of individual waste packages. For waste suspected to contain significant quantities of ^{235}U, or which would be present in a physical form susceptible to self-shielding, an active neutron coincidence counter is used [8, 9], incorporating both thermal and fast mode (with a Cadmium liner) operation, for specific waste types.

It is recognised that the NDA measurement accuracy is better for individual packages than for filled drums, and this has been proved as part of the uncertainty assessment. Although confirmatory measurements are also performed on the package counter, on filled drums, these are only as a confirmatory check. Waste is sentenced by summing the results for individual packages, which are then placed into a 200 *l* drum for despatch to WRACS.

Comprehensive performance assessments were also completed for the NDA systems deployed in the uranium recovery plant, for sentencing ^{235}U – contaminated waste to WRACS. These followed the same approach as described above, for the WRACS NDA systems.

5. Optimisation of plant throughput

In practice, a balance must be sought between setting an acceptance criteria (allowed % discrepancy between the declared and measured ^{235}U mass) which is too wide (in which case there is a higher than necessary risk of drums passing through the system with inconsistencies going undetected) and one which is too narrow (in which case there will be a higher than necessary proportion of falsely rejected drums, due to natural statistical variations in the waste drum properties.

We therefore used the results of the uncertainty assessments, presented in section 4, to optimise the conditions of acceptance into the plant, using the uranium recovery plant as a case-study. The conventional [4] 2σ TMU performance figures equate to expressing the measurement uncertainty at the 95 % confidence level. That is, one has 95 % confidence that the true value lies within the $\pm 2\sigma$ range about the mean measurement result.

By combining the TMU figures for the recovery plant and WRACS NDA systems in quadrature, it is possible to calculate an appropriate upper limit for the ^{235}U content of 200 *l* drums despatched from WRACS. In this way, we have been able to derive an upper limit for sentencing, such that, if a drum was sentenced with a mean measurement value at this limit, there is 95 % confidence that the true value is below the fixed acceptance threshold for WRACS.

As a result of this work, we have reduced the working upper limit for the ^{235}U mass loading, for sentencing from the uranium recovery plant. In practice, the plant sentencing limit has now been set, in accordance with the uncertainty assessments presented above, to be 30 % below the WRACS acceptance limit (before the change, it was set nominally, 20 % below). As a result, the proportion of drums being consigned at the upper limit but rejected at WRACS, has reduced substantially. This provides confidence in the validity of the uncertainty performance assessments, outlined in section 4. It also leads to substantial throughput improvements for WRACS, minimising the inconvenience caused to the consigning plant and WRACS, by the essential follow-up investigations.

6. Review of experience with real waste drums

The results for all the drums processed through WRACS, provide a valuable database for studying the performance of the NDA systems for real waste streams encountered in practice. By comparing the NDA19 and NDA20 results, it is possible to assess the validity of their uncertainty estimates, derived as described in section 4. We analysed the statistical spread observed in the NDA20/NDA19 ratio, for over 200 drums consigned from the 2 plants at Dounreay which sentence the majority of the ^{235}U inventory (dominated by the recovery plant), over an approximately 4 year period since routine operations commenced in 2001.

It is recognised that, in practice, there will be some correlations between the results from NDA19 and NDA20. However, the 2 sets of results can be considered as largely independent, due to the different physics of the 2 systems. For example, the shapes of the positional response profiles for NDA19 and NDA20 are quite different. Furthermore, the particular circumstances which are likely to give rise to over or under – estimation in the ^{235}U mass result, are quite different for NDA19 and NDA20. Consider the case where uranium is shielded within a dense moderator such as a graphite crucible, this may clearly lead to over-estimation in NDA20 due to the additional neutron thermalisation, but under-estimation in NDA19 due to the active material being located on the inner surface.

Figure 4 presents the NDA20/NDA19 ratio as a function of the NDA20 ^{235}U mass result (data obtained from [6]). The mean relative standard deviation is ≈ 53 % (including only those points far above the detection limits, taken as > 5 g, in order to ensure that counting statistics are negligible, and excluding 2 obvious outliers that would otherwise have biased this result). If one considers the individual TMU results for NDA20 and NDA19, of ≈ 27 % and ≈ 50 % (in the case of an unknown matrix) respectively, we would expect an overall standard deviation in the ratio, of ≈ 57%, by adding the two results in quadrature. This is broadly consistent with the observed distribution in the real drum results (53 %). This provides confidence in the validity of the uncertainty estimates, and the methodology and assumptions underpinning the method. Moreover, it shows that the properties of the calibration drums used to assess the performance of NDA19 and NDA20, are broadly representative of the range of real waste streams consigned to WRACS. It is interesting to note that the distribution is approximately uniform, not exhibiting a peaked distribution centred around the mean ratio.

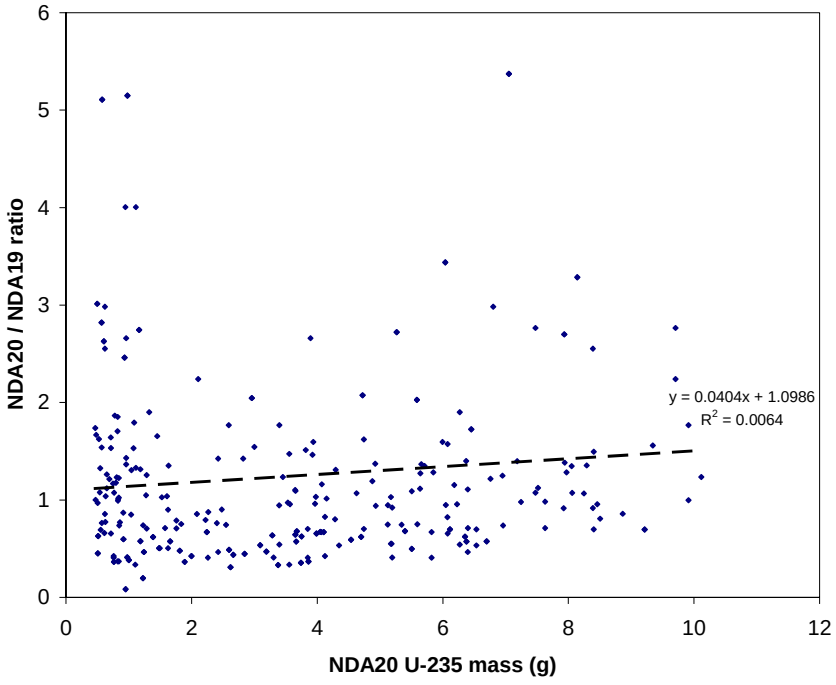

Figure 4. Variation of NDA20/NDA19 reported ^{235}U mass ratio, for approx 200 real waste drums.

7. Conclusions

We have presented a summary of comprehensive uncertainty assessments for a suite of NDA instrumentation, used to consign and check drums of ^{235}U – bearing LLW at Dounreay. These assessments have been performed according to accepted UK industry best practice guidelines. By combining the known uncertainties for the NDA systems used at the point of sentencing, and upon receipt in to the LLW processing plant (for verification purposes) at Dounreay, we have described an approach to modifying the acceptance criteria, with a view to minimising the proportion of drums that are falsely rejected. We have shown that this modification has led to a substantial reduction in the reject proportion, under normal operating conditions. This supports the validity of the uncertainty assessments.

By comparing the results of a large set of real waste drums, for 2 independent NDA systems, both used to verify the ^{235}U inventory of consigned LLW drums, we have demonstrated the validity of the uncertainty estimates. This confirms both that the calibration drums adequately represent the properties of real waste streams, and the validity of the methodology and assumptions underpinning the uncertainty assessments.

These studies highlight the generic benefits of comprehensive surveys and analyses of real drum assay results from multiple assay systems. Studies of this kind can give confidence in the validity of the system calibrations and performance assessments, highlighting recommended areas of future study, as appropriate, to refine the performance assessments.

8. Acknowledgements

Thanks are due to the staff in the archives section at Dounreay, for their help in gathering the historical data on real waste drums.

8. References

[1] M. Dunnet, “Management of Low Level Waste at Dounreay”, Paper presented at EPRI LLW conference, Thurso, United Kingdom, September 2002.

[2] S. Croft, J.P. Argyle, P.M.J. Chard, I.G. Hutchinson, K.P. Lambert, D.J. Lloyd, P.J. Price and L.C. Russen. “An advanced nuclear materials assay system for plutonium contaminated Low Level Wastes”, Proceedings of the 40th Annual Meeting of the Institute of Nuclear Materials Management (INMM), Phoenix, Arizona, 25 - 29 July 1999.

[3] P Chard, P D McClelland and S Croft. “Optimisation of the performance of a Low Level Waste Passive / Active Neutron Interrogation System”, Proceedings of the 43rd Annual INMM meeting, Florida, June 2002.

[4] National Physical Laboratory, Measurement Good Practice Guide No.34 Radiometric Non-Destructive Assay, 1, National Physical Laboratory, Teddington, Middlesex, January 2002

[5] J F Briesmeister (Ed.), “MCNP™ – A general Monte Carlo N-Particle transport code, version 4C”, Los Alamos National Laboratory report LA-13709-M, March 2000.

[6] M Roydhouse, “Assessment of Non-Destructive Assay systems U-235 reported mass data”, MSc thesis, University of Birmingham, September 2004.

[7] P.M.J. Chard and S. Croft. “Self-shielding factors for the Cf-shuffler neutron interrogation technique”, Proceedings of the 17th ESARDA Symposium on Safeguards & Nuclear Material Management, Aachen, Germany, 9-11 May 1995, pp 189 -194. EUR 16290 EN.

[8] S. Croft, P.M.J. Chard, I.G. Hutchinson and D.J. Lloyd. “Design and performance of the Harwell N33 active well coincidence counter”. Proceedings of the 36th Annual Meeting of the Institute of Nuclear Materials Management (INMM), Palm Desert, California, 9 - 12 July 1995.

[9] “The role of NDA in supporting a historical ILW repackaging project”, paper to be published, Proceedings of the 10th International Conference on Environmental Remediation and Radioactive Waste Management (ICEM), Glasgow, Scotland, September 2005.

The use of Neutron Resonance Capture Analysis to determine the elemental and isotopic composition of nuclear material

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Abstract:

The probability for nuclei to capture neutrons reveals sharp peaks, so-called "resonances", which occur at neutron energies specific for each nuclide. These resonances are very suitable to identify and quantify nuclides in objects and materials. They are the basis of an analytical method called "Neutron-Resonance-Capture-Analysis" (NRCA). NRCA is a fully non-destructive method, is applicable to almost all stable isotopes, determines the bulk elemental composition, does not require any sample preparation and results in a negligible residual activity. Up to now NRCA has been mostly applied for archaeological applications. So far we have preferred to determine weight ratios of elements in materials or objects by comparing with standard samples of known elemental and isotopic composition.

In this paper we review the technique and present an improved analyse procedure, which is based on a more methodological approach. The procedure relies on a full Resonance Shape Analysis (RSA) and accounts directly for the neutron self-shielding, multiple scattering, Doppler broadening and instrumental resolution. Therefore, calibration requirements are significantly reduced and the implementation of RSA makes the method applicable to difficult samples with strongly overlapping resonances. We discuss the applicability of the technique in the nuclear field, notably the detection of neutron-absorbing isotopes in nuclear materials in order to quantify impurities in reference materials and to detect the presence of neutron poison in uranium samples.

Keywords: NRCA; NDA; resonances; uranium; gadolinium; neutron poison; enrichment; reference material; nuclear forensic analysis; illicit trafficking; elemental and isotopic composition; nuclear material

1. Introduction

There are three aspects of the neutron capture process, which can be used to analyse the elemental and isotopic composition of objects and materials [1]. The first aspect concerns the radioactivity induced by neutron capture. Usually small samples are taken from an object and irradiated inside a reactor. After a suitable waiting time, the energy and intensity of the gamma-rays following the radioactive

decay are determined with a high-resolution detector to identify and quantify the elements of the sample. This is known as instrumental-neutron-activation-analysis (INAA) [2]. The second aspect concerns the energy and intensity of prompt gamma-rays emitted directly after neutron capture and again detected with a high resolution detector. This method, known as prompt-gamma-neutron-activation-analysis (PGNAA), has found many applications and has already reached a high degree of sophistication [3]. The third aspect of neutron capture relevant to the analysis of objects relates to the resonances occurring in the neutron energy differential capture cross section. The resonances occur at neutron energies typical for each nucleus. They are the basis of an analytical method, "Neutron-Resonance-Capture-Analysis" (NRCA), which has been developed in a joint project of IRI (Delft, NL) and IRMM (Geel, B) [4]. NRCA is a non-destructive method that is applicable to almost all stable isotopes, determines the bulk elemental composition, does not require any sample taking or surface cleaning, and results in a negligible residual radioactivity. We have validated the method by a comparison of the elemental composition of a bronze arrowhead obtained by NRCA with the results from more conventional INAA [5]. In Ref. [6] we compared the performance of NRCA with PGNAA, and showed that they are related and complementary analytical methods. NRCA has been proven useful for the non-destructive quantitative elemental analysis of bronze archaeological artefacts and similar valuable objects [1,4-6]. Apart from this line of research we started to explore applications with other kinds of materials and within other fields of interest, such as material research and in the biomedical field [7]. In this paper we concentrate on the use of NRCA in the nuclear field.

2. Experimental set-up

The NRCA technique has been developed at the neutron Time-of-Flight (TOF) facility GELINA of the Institute for Reference Materials and Measurements (IRMM) in Geel, Belgium. This facility has been designed and built especially for high-resolution neutron cross-section measurements [8]. It is a multi-user TOF - facility, providing a pulsed white neutron source, with a neutron energy range between 1 meV and 20 MeV. The accelerator is mostly operated at 800 Hz and 70 μA average electron current, providing electron pulses ~1 ns wide and with an average electron energy of ~100 MeV. High-energy electrons generate Bremsstrahlung in a uranium target, where high energy neutrons are produced by (γ,n) and (γ,f) reactions. Two water-filled Be containers 4 cm thick are used as moderators to increase the yield of neutrons in the region below ~100 keV. The moderated spectrum shows a Maxwellian distribution plus a tail of partially moderated neutrons with an intensity approximately proportional to the inverse of the neutron energy.

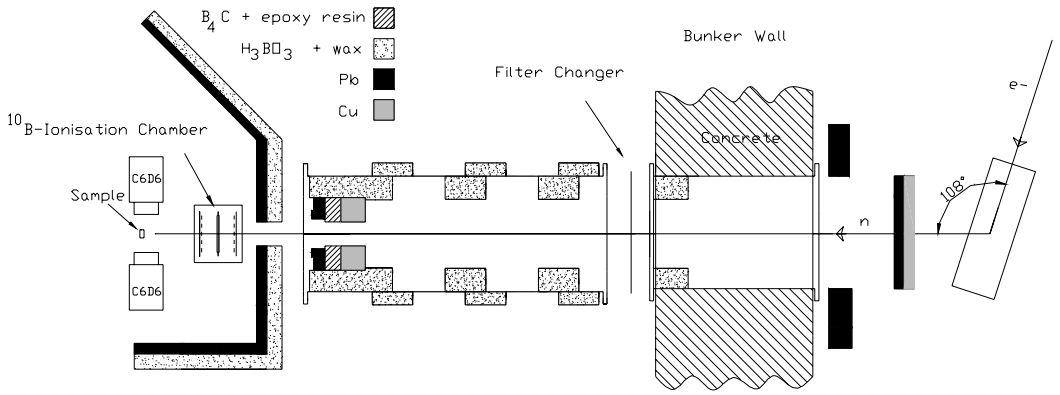

Figure 1: Experimental arrangement of the NRCA measurement station. For more details on the accelerator, neutron target and moderator we refer to Ref.8.

The experimental arrangement is shown in Fig.1. A shadow bar made of Cu and Pb is placed close to the uranium target to reduce the gamma-flash and the fast neutron component. The measurements were performed at a 14.36 m flight-path, forming an angle of 108° with the direction of the electron beam. In normal accelerator operating conditions the neutron flux at the sample position in the centre of the beam is approximately $4000xE^{-0.9}$ per ($cm^2 \cdot sec \cdot eV$). The moderated neutron beam was collimated to ~75 mm in diameter at the sample position. The collimation system was mainly composed of B_4C mixed with epoxy resin, H_3BO_3 mixed with wax, Cu and Pb collimators. Almost halfway between the neutron target and the sample position, i.e. just outside the 3 m thick bunker wall, a sample changer for permanent (^{nat}Cd, Pb and Na) and black resonance filters (Ag, W, and Co) was installed. A 1 mm thick ^{nat}Cd anti-overlap filter eliminated the influence of slow neutrons from a previous accelerator cycle and a 9 mm thick Pb filter was installed to further reduce the intense gamma-flash. A fixed Na filter was used to continuously monitor the background at 2.85 keV. To minimize the influence of nearby flight paths, shielding walls were built around the detectors. In the measurement station an air-conditioning system was installed to keep the sample at a constant temperature and to avoid electronic drifts due to temperature changes.

The shape of the neutron spectrum was measured with a double Frisch-gridded ionization chamber placed 80 cm before the sample. This chamber has a cathode loaded with two back-to-back layers of about 40 $\mu g/cm^2$ ^{10}B each. Two BF_3 proportional counters were used to monitor the stability of the neutron output from the accelerator and there counts are used to normalize the spectra to the same total neutron intensity. The resonance structure can be observed by the time-of-flight (TOF) technique using a set of high efficiency γ-ray detectors. The gamma rays, originating from the neutron capture reaction, were detected by a pair of C_6D_6-based liquid scintillators (NE230) of 10 cm diameter and 7.5 cm length, which were mounted outside the neutron beam at the sample position and oriented perpendicularly to the neutron beam direction. Each scintillator was coupled to an EMI9823-KQB photomultiplier through a quartz window, reducing the neutron sensitivity of the detectors as much as possible. Since the main interaction process in a C_6D_6 detector is the Compton effect and pair production, these detectors have a very poor gamma ray energy resolution. However, C_6D_6 detectors have an excellent time resolution and a low sensitivity to incident neutrons. Therefore, these detectors are optimum detectors for neutron TOF-measurements.

The TOF of a neutron is determined by the time between the start signal, given at each electron burst, and the stop signal from the capture detectors. These pulses are sent to a Fast Time Coder with a 0.5 ns time resolution, designed at the IRMM [9]. The TOF together with the pulse height of the detected events are recorded in list mode using a data acquisition system developed at the IRMM [10]. The list mode allows off-line analysis of the data. The neutron time-of-flight along a flight path defines the neutron velocity and, accordingly, the neutron energy E_n, which in the non-relativistic case is:

$$E_n = \frac{1}{2} m_n \frac{L^2}{T_n^2} \qquad (1)$$

where m_n is the neutron mass, L the distance travelled, and T_n the time of flight.

3. Data analysis

For a regular object the response of the capture system $C_{c,k}$ in the region of a well isolated resonance of a nuclide k can be expressed as a function of the capture yield $Y_{c,k}$, the incoming neutron flux φ and the efficiency to detect the capture event $\varepsilon_{c,k}$:

$$C_{c,k}(E_n) \cong \varepsilon_{c,k}(E_n)\, Y_{c,k}(E_n)\, \varphi(E_n) \qquad (2).$$

The capture yield, i.e. the fraction of the neutron beam that undergoes a (n,γ) reaction in the sample, is a function of the total and capture cross section of nuclide k, expressed as $\sigma_{t,k}$ and $\sigma_{\gamma,k}$, respectively:

$$Y_{c,k}(E_n) = (1 - e^{-\sum_j n_j \sigma_{t,j}(E_n)}) \frac{n_k \sigma_{\gamma,k}(E_n)}{\sum_j n_j \sigma_{t,j}(E_n)} + \mu_k \qquad (3)$$

where n_k is the effective thickness for the nucleus k expressed in atoms/barn. The term in brackets accounts for the self-shielding effect and μ_k for the multiple scattering correction.

For a correct treatment of the self-shielding the total cross-section should be Doppler broadened. The observed line shape is a

convolution of Eq.3 with the time resolution of the spectrometer. The basic information to be extracted from the data needed for a quantitative analysis is the total number of counts in the resonance peaks of interest. For thin objects, weak resonances, or minor components, such that $n_k\sigma_{t,k} << 1$, we can neglect the self-shielding in Eq.3 and the observed resonance area is directly proportional to the amount of material.

So far we have determined the weight ratios of nuclides of the object (or material under study) by comparing with standard samples of known composition. That is, we use the expression:

$$\frac{n_k}{n_m} = A_c \cdot R(E_{r,k};E_{r,m}) \cdot \frac{N(E_{r,k})}{N(E_{r,m})} \qquad (4)$$

The n_k and n_m are the amount of the nuclides k and m, respectively, and $N(E_{r,k})$ and $N(E_{r,m})$ the integrated areas of the resonance. The factor R is the ratio of the self-shielding factors of the resonances at $E_{r,m}$ and $E_{r,k}$. A_c is the calibration factor and is determined by a measurement of the area ratio for the same resonances of a representative calibration sample with known composition. This factor also accounts for changes in the detection efficiency, which can differ from resonance to resonance. For the self-shielding correction the thickness of the object is required, which can be obtained from the ratio of the integrated areas of two resonances with a different strength as discussed in Ref.[11]. This technique requires additional calibration experiments prior to the measurement of the unknown object.

In case of strong resonances and/or thick samples the multiple scattering correction in Eq. 3 cannot be neglected and the calibration approach can not always be applied. Also for samples with overlapping resonances a more sophisticated analysis is required. Such an analysis should be based on a more methodological approach by identifying the main metrological parameters of the measurement process and applying a so-called Resonance Shape Analysis, such as the RSA code REFIT developed by Moxon [12]. This code accounts for the experimental resolution and Doppler broadening, and the self-shielding and multiple scattering in the object.

To perform such an RSA the capture response C_c needs to be transferred into an experimental observed capture yield. Therefore, the response of the capture detector needs to be corrected for the energy dependence of the neutron flux and the detection efficiency for a capture event In a given isotope. These corrections are analogous to the ones that are applied when performing high resolution capture cross section measurements [13,14]. To correct for the shape of the neutron spectrum we use the response of the ^{10}B ionisation chamber. To account for the efficiency to detect a capture event we apply the total energy detection principle in combination with the Pulse Height Weighting Technique (PHWT) [15]. The total energy detection principle is based on the use of a low efficiency detection system, with a gamma-ray detection efficiency that is proportional to the total gamma-ray energy emitted when a neutron is captured in a given isotope [14]. To achieve this proportionality, the PHWT is applied which is based on a mathematical manipulation of the response function of the detection system. One defines a weighting function $W(E_d)$, which satisfies the following relation:

$$\int_0^\infty R(E_d,E_\gamma)W(E_d)dE_d = kE_\gamma \qquad (5),$$

where $R(E_d,E_\gamma)$ the detector response, i.e. the probability that a gamma ray with an energy E_γ results in an energy deposition of energy E_d. It is shown that the weighting function $W(E_d)$ can be approximated by a smooth function of the deposited energy E_d. Under these conditions the detection efficiency for a capture event ε_c is directly proportional to the total energy released in the capture event E_x (the sum of the neutron binding and neutron kinetic energy in the center of mass system):

$$\varepsilon_c(E_n) \approx k\sum_i E_{\gamma i} = kE_x \qquad (6)$$

and independent of the actual cascade path.

Finally, the experimental observed capture yield $Y_{c,exp}$ is derived from the ratio of the weighted response of the capture detection system and the response of the ^{10}B ionization chamber, both corrected for their relative efficiencies:

$$Y_{c,exp} = K\frac{C_{wc}}{C_n}\frac{\sigma_\alpha}{E_x} \qquad (5),$$

where K is an energy independent normalization factor and σ_α is the $^{10}B(n,\alpha)$ cross-section, C_n the count rate from the ^{10}B-ionization chamber, and C_{wc} the weighted count rate of the C_6D_6 detection system, corrected for dead-time losses and background. To derive the capture yield from the raw TOF spectra we used the data processing package AGS (Analysis of Geel Spectra) developed at the IRMM by Bastian

[16]. This package includes the most important spectra manipulations, such as: dead time correction, background subtraction, normalization and TOF to energy conversion. The package includes a full propagation of uncertainties, accounting for both uncorrelated and correlated uncertainty components.

4. Applications

Up to now NRCA has been mostly applied for archaeological applications. The improvements in the data analysis procedures open the possibility to widen the field of applications. In this section we demonstrate the exploitation of NRCA to quantify impurities in reference materials and to detect the presence of neutron poisons, such as gadolinium, in nuclear materials. To determine the nuclide composition of the objects we performed a resonance shape analysis using the REFIT code. The experimental observed capture yield was fitted by a least square procedure by adjusting the relative amount of the observed nuclides and considering all other variables, such as the resonance parameters, as fixed parameters.

4.1. Determination of impurities in reference materials

Recently capture and transmission experiments were carried out at GELINA to improve the experimental data for the ^{103}Rh(n,γ) cross-section in the thermal and the resolved resonance region [18]. Improved capture cross section data are motivated by the objective to extend and optimise the operation of present nuclear power plants and for criticality safety studies of spent fuel storages. The thermal capture cross section is requested with an accuracy better than 2%. To determine the cross section with such accuracy the ^{103}Rh target needs to be very well characterised. We performed NRCA measurements to identify and quantify impurities, which contribute to the response of the capture detection system. The results of the analysis with REFIT are summarised in Table 1. The impurities listed in Table 1 contribute for 0.5 % to the observed count rate of the capture detection system in the thermal region. Therefore, the characterization of the ^{103}Rh target was important for the final analysis of the data. Another example of the use of NRCA for the characterization of nuclear material is given in Ref. [19], were the quantity of O, S and Pb in a lead-iodine sample, originating from reprocessed fuel, was determined.

Nuclide	Relative amount (wt %)	
^{103}Rh	99.5137	
^{181}Ta	0.0337	(0.0029)
^{191}Ir	0.0870	(0.0033)
^{193}Ir	0.1478	(0.0076)
^{182}W	0.0552	(0.0027)
^{183}W	0.0302	(0.0028)
^{186}W	0.0613	(0.0025)
^{197}Au	0.0059	(0.0011)

Table 1: The result of the characterization of the ^{103}Rh metal disc with NRCA.

4.2. Determination of neutron poison in nuclear material

To demonstrate the capabilities of NRCA for the detection of neutron poison in nuclear material we analysed a set of well-characterised mixtures of U_3O_8 and Gd_2O_3 powders. The isotopic composition of the batch of depleted U_3O_8, obtained by mass spectrometry, is given in Table 2. The other relevant characteristics of the samples are reported in Table 3. The amount of ^{155}Gd and ^{157}Gd relative to ^{238}U was deduced from the total weight of Gd_2O_3 and U_3O_8 put into each sample, the isotopic composition of the depleted U and the natural abundances of the Gd isotopes.

In Figure 2 we show the results of an RSA with REFIT for the sample with a 14 wt% Gd/U ratio. The resonance parameters used in the analysis were taken from JEF data files. The NRCA results for the amount of ^{155}Gd and ^{157}Gd relative to ^{238}U are given in Table 3. The uncertainties quoted only account for the counting statistics and do not include any uncertainty resulting from the nuclear data. The NRCA results for n(^{155}Gd) / n(^{238}U) are in very good agreement with the declared data. The largest deviation is less than 0.35 %. The NRCA results for n(^{157}Gd) / n(^{238}U) are systematically higher than the declared value. This systematic difference is larger than the quoted uncertainty and might result from the nuclear data that has been used. Before

drawing any conclusion the difference is subject for further investigation.

Isotope	Atom %
^{234}U	0.002
^{235}U	0.3206
^{236}U	0.0147
^{238}U	99.6627

Table 2 : The isotopic composition of the batch of depleted U_3O_8 powder

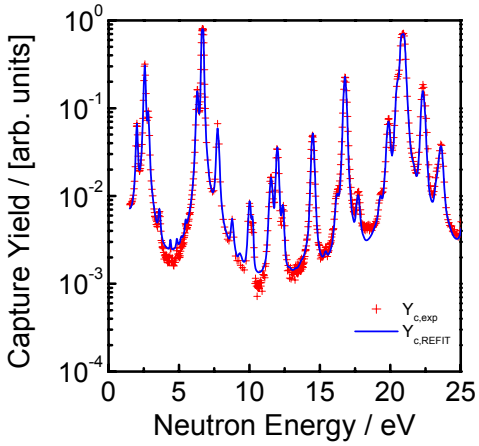

Figure 2 : The capture yield for the mixed U_3O_8 + Gd_2O_3 sample with a 14 wt% Gd/U ratio. The experimental data are compared with the results of an RSA using the REFIT code. The main resonances in ^{238}U are at 6.7 and 20.9 eV. The remaining peaks with peak yields above ~0.005 are due to resonances in the gadolinium isotopes

U / g	Gd / g	Gd/U in wt%	n(^{155}Gd) / n(^{238}U)		n(^{155}Gd) / n(^{238}U)	
			Declared	NRCA	Declared	NRCA
20.988	0.0536	0.255	$5.77\ 10^{-4}$	$(5.76 \pm 0.04)\ 10^{-4}$	$6.10\ 10^{-4}$	$(6.59 \pm 0.07)\ 10^{-4}$
20.608	0.5206	2.530	$5.71\ 10^{-3}$	$(5.73 \pm 0.01)\ 10^{-4}$	$6.03\ 10^{-3}$	$(6.53 \pm 0.02)\ 10^{-3}$
18.656	2.6240	14.065	$3.13\ 10^{-2}$	$(3.14 \pm 0.01)\ 10^{-4}$	$3.36\ 10^{-2}$	$(3.51 \pm 0.03)\ 10^{-2}$

Table 3: The characteristics of the mixed U_3O_8 + Gd_2O_3 powder samples. The isotopic composition of the U_3O_8 powder is given in Table 2. We also compare the NRCA results with the declared data. The uncertainties quoted (one standard deviation) only result from counting statistics.

5. Conclusions

We have demonstrated that NRCA can be used as a radiometric method to determine both the elemental and isotopic composition of materials. NRCA is a non-destructive method with a high sensitivity for nuclides with resonances in the eV-region. NRCA can be used to detect a large number of nuclides. It results in a negligible residual activity and does not produce nuclear waste.

Applying the data analysis procedure presented it this work NRCA can also be applied to complicated samples, resulting in overlapping resonances. In addition, the results deduced from an RSA using REFIT are traceable to nuclear data.

We also showed that NRCA can be applied for the characterisation of reference materials used for cross section measurements and for the determination of neutron poison in nuclear material. Since NRCA is sensitive to a wide range of elements and also determines the isotopic composition, the method could be very useful for nuclear forensic analysis.

6. Acknowledgements

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7. References

[1] Postma H., Schillebeeckx P. and Halbertsma R.B., “Neutron-Resonance-Capture-Analysis of some genuine and fake Eutruscan copper-alloy statuettes”, Archaeometry 46 (2004) 365

[2] De Corte F., Simonits A., Bellemans F., Freitas M.C., Jovanović, Smodiš, Erdtmann G., Petri H. and De Wispelaere A., "Recent advances in the ko- standardization of neutron activation analysis: extensions, applications, prospects", J. of Radioanal. Nucl. Chem., 169 (1993) 125

[3] Molnár G., Belgya T., Dabolczi L., Fazekas B., Révay Zs., Veres A., Bikit I., Kis Z., Ostor J., "The new prompt gamma-activation analysis facility at Budapest", J. Radioanal. Nucl. Chem. 215 (1997)

[4] Postma H., Blaauw M., Bode P., Mutti P., Corvi F., Siegler P., "Neutron-resonance capture analysis of materials", J. of Radioanal. Nucl. Chem., 248 (2001) 115.

[5] Blaauw M., Postma H. and Mutti P., "Quantitative Neutron Resonance Capture Resonance Analysis Verified with Instrumental Neutron Activation Analysis", Nucl. Instr. Meth. A505 (2003) 508

[6] Postma H. and Schillebeeckx P., "Non – destructive analysis of objects using neutron resonance capture", accepted for publication in J. for Radioanal. Nucl. Chem. 2005

[7] Perego R.C., Postma H., Blaauw M., Schillebeeckx P. and Borella A., "Neutron Resonance Capture Analysis: improvements for resonances above 3 keV and new applications", Proc. Modern Trends in Activation Analysis, Guildford, UK, June 20 – 25, 2004

[8] Böckhoff K.H., Carlson A.D., Wasson O.A., Harvey J.A., and Larson D.C., "Electron linear accelerators for fast neutron data measurements in support of fusion energy applications", Nucl. Sci. & Eng., 106 (1990) 192

[9] de Jonge, S., Internal Rep. GE/DE/R/24/87, IRMM, Geel

[10] Gonzalez, J., Bastian, C., de Jonge, S., Hofmans, K., Internal Rep. GE/R/INF/06/97, IRMM, Geel

[11] Postma H., Blaauw M., Schillebeeckx P., Lobo G., Halbertsma R.B. and A.J. Nijboer, "Non-Destructive Elemental Analysis of Copper-Alloy Artefacts with Epithermal Neutron-Resonance Capture", Czech. J. Physics, 33 (2003) A233

[12] Moxon M.C. and Brisland J.B., "GEEL REFIT, A least squares fitting program for resonance analysis of neturon transmission and capture data computer code", AEA-InTec-0630, AEA Technology, October 1991

[13] P. Schillebeeckx, A. Borella, A. Brusegan, J. Heyse, G. Noguere, H. Postma, P. Siegler, K. Volev and W. Mondelaers, "High Resolution Cross-section Measurements at the TOF-facility GELINA", XV International School on Nuclear Physics, Neutron Physics and Nuclear Energy September 9 - 13, 2003 Varna, Bulgaria

[14] Borella A., Volev K., Brusegan A., Schillebeeckx P., Corvi F., Koyumdjieva N., Janeva N. and Lukyanov A., "Determination of the ^{232}Th(n,γ) Cross Section from 4 to 140 keV at GELINA", accepted for publication Nucl. Sci. Eng.

[15] Macklin R.L. and Gibbons J.H., "Capture-Cross-Section Studies for 30-220-KeV Neutron Using a New Technique", Phys. Rev, 159 (1967) 1007

[16] Borella A., Aerts G., Gunsing, F. Moens A., Wynants R. and Schillebeeckx P., "Determination of weighting functions and neutron sensitivity for C_6D_6 detectors By MCNP", Proc. of the International Conference on Nuclear Data for Science & Technology - ND2004, September 26 - October 1, 2004, Eldorado Hotel, Santa Fe, New Mexico

[17] C. Bastian, "AGS, a set of UNIX commands for neutron data reduction", Proc. Int. Conf. On Neutron Research and Industry, Crete, Greece, 1996, Ed. G. Vourvopoulos, SPIE – The International Society for Optical Engineering, Vol. 2867, 611, 1997

[18] Brusegan A., Berthoumieux E., Borella A., Gunsing F., Moxon M., Siegler P. and Schillebeeckx P., "Neutron Capture And Transmission Measurements On ^{103}Rh Down To Thermal Energies ", Proc. of the International Conference on Nuclear Data for Science & Technology - ND2004, September 26 - October 1, 2004, Eldorado Hotel, Santa Fe, New Mexico

[19] Noguerre G., "Mesures et analyses des sections efficaces neutroniques totales et de captures radiatives des iodes 127 et 129 de 0.5 eV à 100 keV ; Effets expérimentaux et études de sensibilité", PhD Thesis, CEA Cadarache, December 2003

Session 16

Additional Protocol – Methodology

Additional Protocol preparation with respect to a supra-national organisation.

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Abstract

Multi disciplinary and multi national research organisation experience with the preparation coordination and review procedures associated with the Additional Protocol Preparation.
We have to balance the guarantee given to the customer and staff working inside the centre that commercial secrecy is maintained whilst giving full support to the NPT inspection ideals.

Keywords: additional protocol; safeguards; declaration; design verification; methodology

1. Introduction

The Joint Research Centre is a research based policy support organisation and an integral part of the European Commission. It provides scientific advice and technical know-how in support of EU policies. The status as a Commission service, guarantees independence from private or national interests, is crucial for pursuing our mission.

Institute / Directorate	Country	Nuclear	Shared site	AP declaration
Ispra Site Directorate	Italy	Yes	No	Yes
ITU – Karlsruhe	Germany	Yes	No but ITU is within FZK	Yes
IRMM – Geel	Belgium	Yes	No	Yes
IE - Petten	Netherlands	Yes	Yes	Yes
IPTS - Seville	Spain	No	No	No
IPSC - Ispra	Italy	No	No	No
IES - Ispra	Italy	No	No	No
IHCP - Ispra	Italy	No	No	No

Table 1. Additional Protocol Information Relative to Joint Research Centre Sites

There are seven different institutes in five countries carrying out extensive research of direct concern to European citizens and industry. Over the years, the JRC has developed special skills and unique tools to provide autonomous and Europe-wide expertise to improve understanding of the links between technology, the economy and society. Activities range from the assessment of safety standards for children's toys and improved biomaterials for hip implants to new technologies for

recycling water and the use of satellite systems to monitor land use and deforestation. Some institutes handle nuclear material while others do not, and some are on shared sites.

This report will document the work performed at the Joint Research Centre (JRC) to provide all design information verification, previously known as Model Protocol Additional to Safeguards Agreements. The JRC due to its status had to be considered as an independent country so it provided a declaration directly to Euratom Safeguards Directorate (ESD) in the same way as all of the other European Countries. However, problems were identified due to the presence of organisations within the site that were owned or operated by companies outside of the privileges and immunities accorded to the European Commission.

2. Problems related to the declaration

It was very difficult to switch from the declaration of nuclear material to the declaration of the items identified in the Additional Protocol. The previous system of checks and verification relied on the monitoring and audit of fissile material and not design information and equipment. This resulted in the initial response from a number of people being “what has this got to do with us”? The explanation then required significant correspondence to fully explain the aims, objectives and requirements of the Additional Protocol. The JRC aimed to be fully transparent and cooperative as well as being in full technical compliance. Nevertheless, the problem is to guarantee that research activity can proceed whilst maintaining commercial secrecy according to Article 7.a. [1].

The Ispra site was set up in 1959 for nuclear research into reactor types and nuclear fuel technology. From 1983, when the ESSOR reactor was finally shut down, to 2005 the site research activities progressively changed from predominantly nuclear to mainly environmental activities. The present operational nuclear activities are as follows:
Safeguards Research – non destructive assay utilising small fissile material samples
Cyclotron – isotope production for medical and industrial usage
Decommissioning of site historical nuclear liabilities including 2 nuclear reactors, nuclear research facilities (Fuel melt And Release Oven (FARO), Radiochemistry, high radiation shielded cell facility (LCSR),
Performance Laboratory (PERLA – present programme is to reduce nuclear activity)

Decommissioning constitutes the largest nuclear component on site and it has required the building of new facilities for radioactive material characterisation, handling and storage.
Due to the number of nuclear activities related to the fuel cycle, a detailed description has been provided for these parts of the site, whereas, brief definitions have been provided for buildings outside of this area. It should be noted that Managed Access criteria has been applied by the JRC in areas not related to the additional protocol in order to guarantee commercial security as far as reasonably practicable.

There were also problems with the homogenisation of the declaration and in the use of the software provided for creation of the document (Site Investigation Tool - SIT) [2].

3. Software Experience

The software used was specifically written in order to support preparation of the Additional Protocol declaration. The software used at the JRC was SIT. The SIT project started in 2000 at JRC Ispra with the objective to develop a number of software tools for supporting Additional Protocol activities. All these software tools are based on low-cost GIS (Geographic Information System) platforms and make use of several other technologies, e.g. remote sensing, GPS, mobile GIS, and Web applications. There are versions of SIT that assist with declaration, verification and inspection.

By using SIT the site layout map could be easily generated and attached to the declaration. The SIT approach offered significant advantages compared with declaration tools not based on site topology. These included a more powerful check on declaration completeness, the automatic generation of references to other articles and query declaration from the site map. SIT being compatible with both the IAEA Protocol Reporter software and with the European CAPE software could also be used as a converter between these two formats.

4. Security Problems

The declaration has been given the security classification of restricted [3]. For this reason all communication containing sensitive information has to be performed in a secure way. Information exchange via e-mail should not be different and should only be performed using some kind of 'Secure E-mail' solution using both encryption and digital signatures. ESO suggested the use of PGP whereas the European Commission follows a different 'public/private' key technical solution well integrated with its Reference Configuration and Services like the e-mail service.

The JRC Security Service did not consider the use of PGP as it would be difficult to integrate within the existing working environment and difficult to support. Additionally, it would have been necessary to establish new procedures for the secure generation, exchange and diffusion of involved users' public keys. All these procedures and organisational issues had already been developed and defined according to strict and common security standards and practices within “Commissign” and SECEM ONE (SECure E-Mail based on Outlook Native Encryption) projects so it was decided to follow these consolidated, tested and reliable procedures. Therefore, all JRC internal communication containing sensitive information relative to the additional protocol was transmitted using the European Commission SECEM ONE. It should be noted that interoperability of the Commission's solution has been also thoroughly tested and proved.

The final declaration was transmitted directly onto the ESO server using the CIRCA data transfer system as advised by Euratom Inspectors.

The JRC still expresses reservation about the security of the information once it has passed into the IAEA system.

5. Requirement for Transparency

DG JRC has aimed to be as transparent as possible for the Declaration and Inspection procedures and provide all information required when that is not covered by a commercial secrecy agreement. Any lack of transparency would undermine the ability of Safeguards to ensure that DG JRC is in full compliance with its Non Proliferation Treaty agreements. Information has been provided on a need to know basis that is sufficient for the DG JRC to show full NPT compliance to the IAEA.

6. Political Support for Verification Activities is Imperative

As the JRC is part of the same organization as the European Inspectorate for Safeguards, it is considered an imperative to be seen to fulfill the requirement, to be transparent, cooperative and to suggest improvements as provided in the past for traditional Safeguards. DG JRC has provided training for Euratom Safeguards and it has been proposed to use the Ispra JRC site for future IAEA AP Inspector training.

7. Regulation of the Inspection by DG JRC

The following items were identified during meetings with DG TREN and other JRC sites in order to limit the scope of the NPT inspection exercise.

- Any equipment or material defined in Annex 2 of the Declaration must be labeled with the following instructions, “This item must not be moved without informing the Additional Protocol Site Representative.” If the item is moved another declaration must be written. However, if the item is not included in Annex 2 but is considered to be within the spirit of the additional protocol it must be subject to the same rule.

- In principle, no photographs can be taken if a building or process is identified as MANAGED ACCESS. However, permission can be requested from the Site Representative on an individual photograph basis.

- DG JRC does not have any right to stop the inspector from taking wipe samples but the exact position may be photographed in order to perform a comparative analysis.

- The inspector cannot stop an operational activity that is in progress e.g. chemical process.

- The JRC must review all photographs before the inspection team leave the site and check for relevance to the inspection.

- The JRC must receive a copy of all photographs taken by the inspection team. In ITU the photographs were taken by a security officer and inspectors only received a hard copy instead of an electronic copy.

- The JRC must declare the working times and holidays on each site to avoid inspections outside normal working periods.

- The JRC specified the security classification of buildings, areas and processes visited by the inspection team. Buildings – Managed access, information - public, unclassified, commercial, restricted, secret etc. The Inspection team and the IAEA must then handle the information taken according to the security classification.

- If the JRC identifies a photograph as being inappropriate, the JRC can request that this photograph is deleted.

- For ease of administration each JRC site had a dedicated e-mail box ADDITTIONAL PROTOCOL.

- Only information strictly relevant to the additional protocol was provided. Other information for example concerning security or commercial matters was deemed as being outside the remit of the inspection and was not provided.

- Long descriptions were referenced as Word documents and attached to the individual declaration entry within the SIT software.

To facilitate ease of access for the inspectors it was decided to identify dates and times that would be particularly unsuitable. Information was provided indicating normal working times as well as national and commission holidays. Additionally, it is Commission policy for new workers at the JRC to arrive on the 1st and 15th of every month so queues at the main entrance will be longer for all visitors on these dates. To avoid long entrance delays inspections should avoid these dates.

Name or State should have been DG JRC but the software did not allow this as an option.

Security Classification required and Managed Access was declared. e.g. cyclotron for reasons of safety, security and commercial (Amersham); all ITU wings in particular for safety and security reasons.

Periodically checks are required that the security level of the inspector complies with DG JRC internal security rules.

The following items were identified as being of particular interest to JRC ITU Karlsruhe

During the first Additional Protocol Inspection at the Trans Uranium Institute (ITU) the IAEA inspector stated that ITU would follow German State Regulations. There were some differences with ITU inspection regulation and that of other JRC sites.
These included:

- Photographs could only be take whilst in the presence and with the approval of the security officer and the laboratory manager.
- The inspectors requested non classified documentation available to the public.
- The inspection took 2 hours.
- Verification of use of laboratories and offices took place.
- No request was made to check documentation or design information
- Environmental samples were taken
- The inspection was described as a design information verification

8. Conclusions

DG JRC has successfully provided its initial declarations with the respect to the Additional Protocol utilizing proprietary software. Some problems were encountered due to the many sites of the JRC being in different countries as each site was required to comply with its own national legislation with respect to appropriate areas, safety and security. A delicate balance has had to be made between maintaining guarantees, for example for safety and secrecy, for the JRC stakeholders whilst giving full support to the NPT inspection. The SIT software was designed within the JRC and therefore has the possibility of being developed in a real NPT inspection environment.

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References

[1.] INFCIRC/540 (Corrected) Model Protocol Additional to the Agreement(s) between State(s) and the IAEA for the Application of Safeguards.
[2.] Poucet, S.Contini, F.Bellezza, SIT-D, A Software Tool to Support Declarations in the Context of the Additional Protocol, ESARDA Conference, Luxembourg, 2003.
[3.] Commissign - European Commission Public Key Infrastructure. Further information at http://europa.eu.int/comm/dgs/personnel_administration/commissign/index_en.htm

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Regime Change?[1]

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Abstract

Following the 1998 nuclear tests in South Asia and later reinforced by revelations about North Korean and Iranian nuclear activities, there has been growing concern about increasing proliferation dangers. At the same time, the prospects of radiological/nuclear terrorism are seen to be rising—since 9/11, concern over a proliferation/terrorism nexus has never been higher. In the face of these growing dangers, the regime is under pressure and there are urgent calls for stronger measures to strengthen the current international nuclear nonproliferation regime, including recommendations to ban additional civilian processing of weapon-useable material or to place this activity under multinational control. As well, there are calls for entirely new tools. As proliferation and terrorism concerns grow, there is a temptation to consider more fundamental changes to the regime. In this context, this paper will address the following: By choice or necessity, will we need to replace the regime centered on the Treaty on the Nonproliferation of Nuclear Weapons (NPT) and the International Atomic Energy Agency (IAEA)? What improvements could ensure it will be the foundation for proliferation resistance as nuclear power continues to grow? What will make it a viable centerpiece of future nonproliferation efforts?

Introduction

Following the 1998 nuclear tests in South Asia and later reinforced by revelations about North Korean and Iranian nuclear activities, there has been growing concern about increasing proliferation dangers. At the same time, the prospects of radiological/nuclear terrorism are seen to be rising—since 9/11, concern over a proliferation/terrorism nexus has never been higher. In the face of these growing dangers, the regime is under pressure and there are urgent calls for stronger measures to strengthen the current international nuclear nonproliferation regime, including recommendations to ban additional civilian processing of weapon-useable material or to place this activity under multinational control. As well, there are calls for entirely new tools. As proliferation and terrorism concerns grow, the temptation to consider fundamental changes to the regime, which in recent years has been voiced at a level not heard before, is understandable. In the leadup to the Review Conference (RevCon) of the Treaty on the Nonproliferation of Nuclear Weapons (NPT), many observers argued that the treaty regime is in crisis and that it might collapse. Are these reasoned views? If the NPT were to collapse, or significantly erode, would it be possible to create something better? Current institutions and treaties still command significant international support, and the existing regime, with all its flaws, probably cannot be replaced without tremendous political and other costs. The more promising approach may be to modify the regime rather than construct a new one. In this context, this paper will address the following: By choice or necessity, will we need to replace the regime centered on the NPT and the International Atomic Energy Agency (IAEA)? What improvements could ensure it will be the foundation for proliferation resistance as nuclear power continues to grow? What will make it a viable centerpiece of future nonproliferation efforts?

Proliferation and Terrorism Threats

The proliferation problem today is serious. There has been growing concern about increasing proliferation dangers, including rogue states; cooperation on weapons of mass destruction (WMD) among rogues; technology diffusion via the Internet as well as loose nukes, materials leakage, brain drain in the former Soviet Union, Pakistan and other states and non-state actors like the A. Q. Khan network; and problems with export controls. In the same vein, the prospects of radiological/nuclear terrorism are seen to be rising—concern over a proliferation/terrorism nexus after 9/11 has never been higher.

Against these problems, one can point to some successes—regime change in Iraq, Libya's commitment and actions to disarm, the disruption of the A.Q. Khan network, wide support for President Bush's Proliferation Security Initiative (PSI) and the like.

Pressures on the Regime

Nonproliferation efforts are under increasing pressure in the face of these challenges, some of which are unprecedented.

The NPT, the centerpiece of the regime, is challenged by:

- non-parties and rogue states acquiring weapons, which cannot be accommodated within the Treaty and which may affect the views of key states such as Japan and Brazil;
- North Korea's withdrawal from the Treaty;
- Iranian programs that raise the troubling issue of noncompliance with the Treaty's provisions but, beyond that, demonstrate the Article IV "loophole";
- limited consensus on compliance enforcement; and
- the emergence of nonstate actors, including black marketeers and possibly terrorists.

The IAEA is also under challenge. The IAEA is restricted by the limits of its verification mandate and burdened by noncompliance issues, which raise questions about the value and effectiveness of international safeguards. The Additional Protocol is an important new tool and should become the standard for safeguards and a condition for nuclear supply. Although three-quarters of states with significant nuclear activities have now brought the Additional Protocol into force, there remain a large number of states that have not yet ratified the Additional Protocol. The Agency and member states are trying to remedy this situation.

The Nuclear Suppliers Group (NSG) rules need to be reinforced and strengthened. There is reason to be concerned about the exports of some NSG states. Technology diffusion, black markets and lateral proliferation also raise questions about the long-term relevance of the NSG.

A review of the UN Security Council (UNSC) reveals limited consensus on regime enforcement. The Security Council was paralyzed on Iraq and North Korea. How would it respond to the crisis in Iran if this issue were brought before it?

"Shocks" to the Regime

Beyond such regime problems, there may be "shocks" to the regime, which may or may not appear likely today. These may include:

- nuclear use in South Asia;
- Pakistani loose nukes and brain drain (beyond A.Q. Khan);
- an Islamist regime in Pakistan;
- nuclear-weapon tests in South Asia or North Korea;
- sale of fissile material or nuclear weapons by North Korea; or
- a preemptive strike on Iranian facilities by Israel.

There may be other such shocks that we do not even have on our radar screens.

Regime Change?

All of these problems and issues pose real threats to the regime and, if they are not addressed, may portend its collapse or increasing irrelevance. It would be foolish to act as if these problems did not exist, or that they could be adequately addressed using exclusively old measures and approaches, or resolved merely by muddling through. In the face of these developments, do we need regime change—a change in the NPT/IAEA regime? It seems the calls for regime change, which in recent years have been voiced at a level not heard before, are rational. They are certainly understandable. But there is reason to be skeptical of the proponents of regime change, as there are reasons to doubt predictions of the regime's imminent demise.

In any event, is it possible to create something better? This is not clear. As noted, current institutions and treaties still command significant international support and consensus—such consensus as exists. But consensus is largely limited on difficult cases and tough issues, especially if coercive measures need to be considered. This suggests little prospect that entirely new institutions would more effectively deal with these issues.

The Prospect of Reform

As opposed to putting forward alternatives to the old regime, the regime must be reformed. In this vein, the President's February 11, 2004 landmark nonproliferation speech put forward initiatives to, inter alia:

- reinterpret Article IV of the NPT to close the loophole that allows proliferants to develop the means for weapons legally;
- take practical steps to limit fuel cycle risks by a ban on new states from developing reprocessing and enrichment technologies;
- strengthen export controls, especially for sensitive technologies;
- deal with noncompliance in part through reforming the IAEA's Board of Governors (BOG); and
- build on successful Cooperative Threat Reduction (CTR) and PSI actions.

Let us consider these initiatives in greater detail, with an understanding that they are put forward in the context of reforming the regime, not replacing it. The proposals do include extra-regime initiatives like PSI, but they are designed to strengthen the regime.

Plugging the NPT Loophole

In his nonproliferation speech of February 11th, the President declared: "The world must create a safe, orderly system to field civilian nuclear plants without adding to the danger of weapons proliferation. … Proliferators must not be allowed to cynically manipulate the NPT to acquire the material and infrastructure necessary for manufacturing illegal weapons."

The problem the President proposed to solve is clear. Article IV of the NPT allows states to develop virtual weapon capabilities by pursuing nuclear enrichment and reprocessing under the cover of peaceful programs. This problem was highlighted by recent revelations concerning Iran, but it is an issue that goes beyond Iran. The President did not call for an amendment to the NPT, which is in practice impossible. Nor did he call for a global ban on reprocessing and enrichment, which would also have been impracticable. He proposed to deal with the problem through a ban on sensitive technologies to states that do not possess them and through export controls.

President Bush specifically proposed that the members of the NSG should:

- refuse to sell enrichment or reprocessing equipment or technology to any state that does not already possess full-scale, functioning enrichment or reprocessing plants;
- allow only states that have signed the Additional Protocol, which requires states to declare a broad range of nuclear activities and facilities, to import nuclear or nuclear-related dual use equipment; and
- ensure that states which renounce enrichment and reprocessing technologies have reliable access, at reasonable cost, to fuel for civilian reactors.

These proposals have evolved. The most controversial is the US proposed ban on reprocessing and enrichment, which is viewed by some states as discriminatory. In any event, it is not at present likely to be agreed. The Canadians have proposed a moratorium and the G-8 Summit at Sea Island, Georgia, in June 2004 effectively created a one-year moratorium. The moratorium may be renewed. IAEA Director-General Mohammed ElBaradei has proposed a five-year moratorium to allow for multinational/multilateral solutions to be negotiated, but this may be too long to be acceptable to many key states.

The Additional Protocol (AP) as a condition of supply has been generally supported in the NSG, but several states are opposed. As well, the G-8 put these forward as an essential standard for safeguards; details will need to be worked out.

Assured supply was put forward as the "carrot" to the export control "sticks" in President Bush's speech. Assured fuel supply involves a collective obligation to states. It must convince the recipient it can count on supply, and that there will be no introduction of extraneous political factors into a decision to provide the fuel. This will be a major challenge.

Reforming the IAEA Board

The Bush Administration has been highlighting the importance of compliance with nonproliferation treaties and agreements. In the context of broad support for the International Atomic Energy Agency—including ensuring that the IAEA has all the tools it needs to fulfill its essential mandate and supporting the Additional Protocol--the President called for:

- the IAEA BOG to create a special committee on safeguards and verification; and
- a prohibition on any state under investigation for proliferation violations to serve or continue serving on the IAEA BOG or on the new special committee.

In the proposal, the special committee of the IAEA Board is to be "made up of governments in good standing with the IAEA." The objective is to ensure the "IAEA is organized to take action when action is required." Does this idea have a reasonable chance of success? Given the recent cases of noncompliance, there is probably no better time for the United States to pursue the creation of a Board committee that "will strengthen the capability of the IAEA to ensure that nations comply with their international obligations." The institutional risks of not acting may be unacceptably great for the Agency, and most key member states should understand that reality. Nonetheless, the prospects will depend, in large part, on the details of the committee. Who will participate? How will this committee function? What additional powers should it have? How would it be staffed?

The President also argued that: "No state under investigation for proliferation violations should be allowed to serve on the IAEA BOG—or on the new special committee. And any state currently on the Board that comes under investigation should be suspended from the Board." The reasoning was clear. According to President Bush: "Allowing potential violators to serve on the Board creates an unacceptable barrier to effective action. The integrity and mission of the IAEA depends on this simple principle: Those actively breaking the rules should not be entrusted with enforcing the rules."

The feasibility of this aspect of the President's proposal hinge on the following questions: Is it possible to prevent a state from being selected to the Board, or to remove one from the Board? Will this be acceptable?

Expanding CTR

The President proposed "to expand our efforts to keep weapons from the Cold War and other dangerous materials out of the wrong hands." Cooperative Threat Reduction efforts with the former Soviet Union are the basis for moving ahead. These efforts involve "dismantling, destroying and securing weapons and materials left over from the Soviet WMD arsenal," In addition, there is an effort to assist "former Soviet states find productive employment for former weapons scientists." While this step builds on the G-8 Global Partnership cooperation, it would expand its scope via additional recipients and additional activities. It also does so by seeking more participants and funds in the cooperative effort.

More specifically, the President called for expanding cooperative efforts beyond the former Soviet Union. "We should expand this cooperation elsewhere in the world. The Partnership originally provided $20 billion in nonproliferation assistance to the former Soviet Union, it should now also work to reduce and secure dangerous materials elsewhere in the world. We will retrain WMD scientists and technicians in countries like Iraq and Libya. We will help nations end the use of weapons-grade uranium in research reactors." The President explicitly referred to assistance to the FSU, Iraq and Libya. But the list appears illustrative rather than exhaustive. There is an opportunity to expand cooperation with other states.

In addition to expanding CTR-style programs to new states, there is also a commitment to new activities. Beyond the CTR activities carried out in cooperation with Russia and other Soviet successor states, the President tied US efforts in the program to reduce use of highly enriched uranium in research reactors with

this step. Here too the activities identified in the speech appear illustrative rather than exhaustive. What other new activities may be contemplated? The Global Threat Reduction Initiative—a US initiative to secure, remove, or dispose of an even broader range of nuclear radiological materials around the world that are vulnerable to theft—is clearly an effort to realize this element of President Bush's vision. Lessons learned in Iraq and Libya may be useful in developing a list of additional activities.

Expanding the PSI

The Proliferation Security Initiative, announced by President Bush in May 2003, is one response to the growing global threat from the proliferation of weapons of mass destruction, delivery systems and related materials. Specifically, it is an effort to enforce compliance with export control regimes that are "gentlemen's' agreements." PSI participants are committed to taking steps to interdict shipments of WMD, delivery systems and related materials at sea, in the air or on land. PSI focuses on rogue states and terrorists.

President Bush declared: "I propose that the work of the PSI be expanded to address more than shipments and transfers. Building on the tools we've developed to fight terrorists, we can take direct action against proliferation networks. We need greater cooperation not just among intelligence and military services, but in law enforcement, as well. PSI participants and other willing nations should use the Interpol and all other means to bring to justice those who traffic in deadly weapons, to shut down their labs, to seize their materials, to freeze their assets."

The heart of what the President's proposed is that "participants in the PSI and other willing nations expand their focus and use Interpol and other mechanisms for law enforcement cooperation to take additional actions to pursue proliferators and end their operations." As President Bush notes, this step builds on counter terrorism cooperation and on the PSI. It will be facilitated by the passage of UN Security Resolution 1540, which was also one of the President's initiatives, and its rapid and effective implementation.

The United States is working with the G-8, the NSG and others to realize these initiatives. In addition to the US proposals, proposals by Director-General ElBaradei and others will be discussed in the months and years to come.

Conclusions

The international nuclear nonproliferation regime was created in a different time to deal with different threats. It has been the basis of international consensus and, for all its problems, it is likely to be with us as we think about dealing with today's and tomorrow's proliferation problems. With reforms, the regime can provide the foundation for all efforts to find institutional means to enhance proliferation resistance. Efforts to strengthen it along lines of the President's February 11th initiatives and former Energy Secretary Spencer Abraham's Global Threat Reduction Initiative are critical if the regime is to meet the challenges of the future.

However, these are important steps, but not a panacea. The United States will need to continue to work with others to strengthen the regime and will need to deal with difficult cases with available means. It is imperative to ensure the regime is not further eroded, and that new difficult cases do not emerge out of flaws in regime.

The states with an interest in the existing regime—particularly the United States—must recognize regime problems and manage them. And the United States and other like-minded states must offer the initiatives and build the consensus necessary to address increasing challenges.

End Note

[1] The views and opinions of the authors expressed herein do not necessarily state or reflect those of The Regents of the University of California, the United States Government, or any agency thereof.

Session 17

Containment and Surveillance

Wireless Networking and its Application in Nuclear Safeguards

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Abstract:

Wireless networking can provide a cost effective and convenient method for installing and operating an unattended or remote monitoring system in an established facility. There is concern, however, that wireless devices can interfere with each other and with other radio systems within the facility. Additionally, there is concern that these devices add a potential risk to the security of the network. Since all data is transmitted in the air, it is possible for an unauthorized user to intercept the data transmissions and/or insert data onto the network if proper security is not in place. This paper will describe a study being undertaken to highlight the benefits of wireless networking, evaluate interference and methods for mitigation, and recommend security architectures. Furthermore, this paper will present results of a demonstration that links various wireless sensors at Sandia National Laboratories (SNL) and the Joint Research Centre, Ispra, (JRC) to each other and in the future, to a laboratory at the International Atomic Energy Agency (IAEA).

Keywords: wireless; Bluetooth; Wi-Fi; remote monitoring

1. Introduction

Installing cabling for remote and/or unattended monitoring in nuclear facilities in which the needed infrastructure doesn't exist can be costly and intrusive to operations. In addition, cabling can be difficult for inspectors to locate and access during inspections. With wireless networks, installation, maintenance, and inspection activities are confined to the devices only. Another benefit to wireless technology is its mobility since cabling does not restrict the sensors' locations.

Two important concerns with using wireless technology are interference and security. Interference occurs when two or more devices within certain proximity transmit at the same time on the same frequency channel. The International Telecommunication Union (ITU) coordinates the use of the frequency spectrum among governments. The Federal Communications Commission (FCC) allocates and regulates frequencies in the United States. Licensing through these agencies allows users to "own" a part of the spectrum thus eliminating interference. There are, however, bands in the spectrum that are unlicensed, known as the ISM bands (two of which are 2.4 GHz and 5 GHz). These bands are available for any device to use, subject to power constraints. Cordless phones and microwave ovens are two common household devices that operate in the ISM bands. 802.11 (Wi-Fi) and Bluetooth are two wireless networking devices that also operate in the ISM bands. As such, the coexistence of these devices within a small area may cause interference. There are ways to reduce the interference, such as separating the devices by a certain distance, using directional antennas,

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employing collision avoidance algorithms, changing channels within the band, and operating in different ISM bands.

The second concern for wireless networking is security. Without security, an adversary with proper equipment can intercept and read the data, alter data, and impersonate an authorized wireless device. The first condition can be remedied by use of encryption, second by data integrity, and the third by user authentication. Each wireless connection should incorporate each of the three security measures.

2. Background

This paper will review in some detail two different prominent wireless technologies: 802.15 (Bluetooth) and 802.11 (Wi-Fi). A third wireless technology, 802.16, or Wi-MAX, will be briefly mentioned because of its potential in the near future. The choice of technology depends on factors such as the range and data rate required.

Bluetooth [1], the more common name for IEEE standards 802.15.1 and 802.15.2 (released in 11/2003), is a short-range cable replacement technology. Although there are advantages to the newer version, most devices still use 802.15.1. There are many applications, including connections between PC's, printers, laptops, mobile phones, and digital cameras. The following lists further technical specifications.

- Frequency: unlicensed 2.4 GHz
- Data rate: 720 kbps
- Power consumption: Class 1 – 100 mW, Class 2 – 2.5 mW, Class 3 – 1 mW
- Range: Class 1 – about 100 m, Class 2 – about 10 m, Class 3 – less than 10 m
- Interference: can interfere with Wi-Fi, but less susceptible due to the number of hops/second and algorithm for collision avoidance in version 2
- Use of spectrum: Frequency Hopping Spread Spectrum (FHSS) at 1600 hops/second
- Some costs: USB Bluetooth adapter $50, RS232 adapter $150

Bluetooth devices "pair" to authenticate each other. The procedure begins with one device selecting a random number and transmitting it to the other device. Next, both devices create an initialization key based on the MAC address of the device that received the random number, the random number itself, and a mutual PIN. The communications during the initialization key phase are sent in the clear. The PIN is between 8 and 128 bits and is selected by the users of the devices wishing to communicate. Next the devices select different random numbers as the previous step and send these to each other encrypted using the initialization key. The devices compute a link key as a function of the two random numbers. The link key allows encrypted and authenticated communications to commence.

Bluetooth can operate in one of three security modes: mode 1 is no security, mode 2 is application level security, and mode 3 is link-layer and includes authentication, authorization, and encryption (described above). Published vulnerabilities of Bluetooth [2] include the inability to secure devices (some devices offer no configuration options), weak key utilization, and no protection from replay attacks. Also, since the initialization key is sent in the clear, if an attacker can determine the initialization key he or she can compute the link key. Some recommendations (from reference 2) for security implementation are: disable discovery mode (allows other devices to find it and access its list of services) on all devices, disable unnecessary profiles and services, limit power to what is needed for an application, only pair devices in a secure environment (to prevent interception of the initialization key), use mode 3 security, select long PINs (at least 64 bits), avoid weak PINs (123456), and avoid devices with fixed PINs. As discussed later in this paper, devices that allow limited pairing are the most secure.

IEEE standard 802.11 [3], [4], [5] also known as Wireless Fidelity (Wi-Fi), is the wireless Ethernet. It can be used in place of wired LANs or for Internet access at designated "hot spots" (such as coffee shops, hotels, and airports). There are three different versions of 802.11: a, b, and g. Additional specifications are listed below.

- Frequency: unlicensed 2.4 GHz for 802.11b, g; unlicensed 5 GHz for 802.11a

- Data rate: 54 Mbps (theory), 20 – 25 Mbps (actual) for 802.11a; 11 Mbps (theory), 2 – 3 Mbps (actual) for 802.11b; 54 Mbps (theory), 10 – 15 Mbps (actual) for 802.11g
- Power consumption: 1W
- Range: 70m with regular antenna, 400 – 500m with improved antenna
- Interference: can interfere with version 1 of Bluetooth
- Use of spectrum: direct sequence spread spectrum (DSSS)
- Some costs: PC card $40 - $80, Routers and Access Points $60 - $130

The 802.11 standard includes security mechanisms known as Wired Equivalent Privacy (WEP) and Wi-Fi Protected Access (WPA). WEP is intended to provide functionality equivalent to physical security inherent in a wired medium. Unfortunately, because of key reuse in the algorithm, an attacker can intercept traffic and eventually crack the keys [6]. Therefore, WEP should not be used as the primary method for security. WPA is intended to fix the problems associated with WEP. WPA uses a better implementation of an encryption algorithm and is not subject to the same key reuse problems of WEP. Virtual Private Networks (VPNs) can provide additional security to the 802.11 networks.

Bluetooth and 802.11b and g all use the 2.4 GHz band. If the 802.11 stations are relatively close to a transmitting Bluetooth device, interference may occur. Bluetooth can use almost the entire bandwidth (79 of the 83.5 available channels), hopping at 1600 hops/second, thus hopping on top of all 802.11 transmissions. 802.11 uses just a third of the 2.4 GHz band (20 MHz). Version 1 of Bluetooth does not listen for 802.11 transmissions and thus does not employ any collision detection or avoidance algorithms. Therefore, it is up to 802.11 to regulate the traffic. 802.11 lowers its data rate and retransmits frames if the receiving station doesn't send back an acknowledgement. Because of these potential interference issues, Bluetooth version 2 uses Adaptive Frequency Hopping (AFH) [7] to avoid collisions. AFH allows Bluetooth to adapt to the environment by detecting fixed sources of interference and excluding those channels deemed "bad". It is important to note that AFH avoids collisions with 802.11 but not with other frequency hopping devices such as 2.4 GHz cordless phones. Figure 1 shows Bluetooth (black) and 802.11 (gray) co-existing in the 2.4 GHz region of the spectrum. This would be the situation without AFH. With AFH, however, Bluetooth will avoid the bands in use by 802.11 (gray).

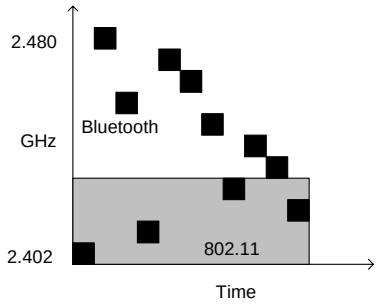


Figure 1: Bluetooth version 1 and 802.11 use of the 2.4 GHz region of the spectrum.

Besides AFH, another alternative is to use 5 GHz 802.11a. Another solution is to use directional antennas to confine power within certain directions.

IEEE 802.16 [8], or Wi-MAX (Worldwide Interoperability for Microwave Access), is a new wireless standard that allows communications between greater distances than Wi-Fi. Whereas Wi-Fi is intended for local area networks (LANs), Wi-MAX is intended for metropolitan area networks (MANs). It can operate with line-of-sight in the 10 – 66 GHz frequencies, and without line-of-sight in the 2 – 11 GHz frequencies (overall using both licensed and unlicensed frequencies). Up to 70 Mbps can be shared in a radius of 50 km. Wi-MAX should begin emerging in cities soon.

3. JRC/SNL Collaboration

Under the auspices of a formal agreement between the United States Department of Energy/National Nuclear Security Administration and the European Atomic Energy Community represented by

Transport and Energy and JRC, SNL and JRC have undertaken a collaborative effort in the development of a remote monitoring system that wirelessly collects sensor data and allows authorized users (whether on a LAN or mobile) to securely access the data. Specifically, JRC and SNL have each developed a remote monitoring system that utilizes Wi-Fi and Bluetooth, respectively, to transfer data to a server connected to the Internet.

The sensor network at SNL consists of a DCM-14 connected to a Win2000 laptop server via Bluetooth. The server periodically "polls" the DCM-14 to collect images that are then stored on the server. The server is connected to a Cisco PIX 501 IPSec VPN and subsequently to the Internet. Figure 2 illustrates both the sensor network architectures for SNL and JRC.

The sensor network at JRC consists of an AXIS Network Camera wired to a Wi-Fi 802.11g Ethernet Bridge, a USB camera connected to a JRC Tablet Server (Win2000), a Wi-Fi 802.11g Access Point for the wireless devices connection and a JRC Server PC connected to the NetScreen 5XT IPSec VPN, which grants the access to the public network (i.e. Internet).

Experiment 1 – **Videoconference** (Figure 3.a):
A user at the SNL server requests a videoconference session with JRC. The request is passed to the SNL VPN gateway, the VPN gateway encrypts all the data packets and redirects them through the Internet to the JRC VPN gateway, which decrypts and forwards the data to the JRC Tablet Server via the Wi-Fi Access Point. The user on the Tablet Server accepts the videoconference request and grants the SNL user access to real time images and video from the USB camera, file transfer, chat, and sharing computer applications and desktops.

Experiment 2 – **Live Remote Surveillance** (Figure 3.b):
Users from the server at SNL request the web address of the AXIS Server Network Camera website, the VPN gateway at SNL encrypts it and forwards across the Internet, the JRC VPN gateway decrypts it and passes it through the Wi-Fi connection to the AXIS Server Network Camera that generates the HTTP response. All response data are encrypted by the JRC VPN gateway, forwarded through the Internet, decrypted by the SNL VPN gateway for the SNL users to view live images from the AXIS Camera.

Experiment 3 – **Future inspection** (Figure 4.a):
Images from the AXIS Camera are stored every 5 seconds (custom setting) on the JRC Server PC. These images are available for future inspection: directly for JRC VPN users and/or from the remote secure connection from SNL VPN users. A dedicated web site is always online on the JRC server PC.

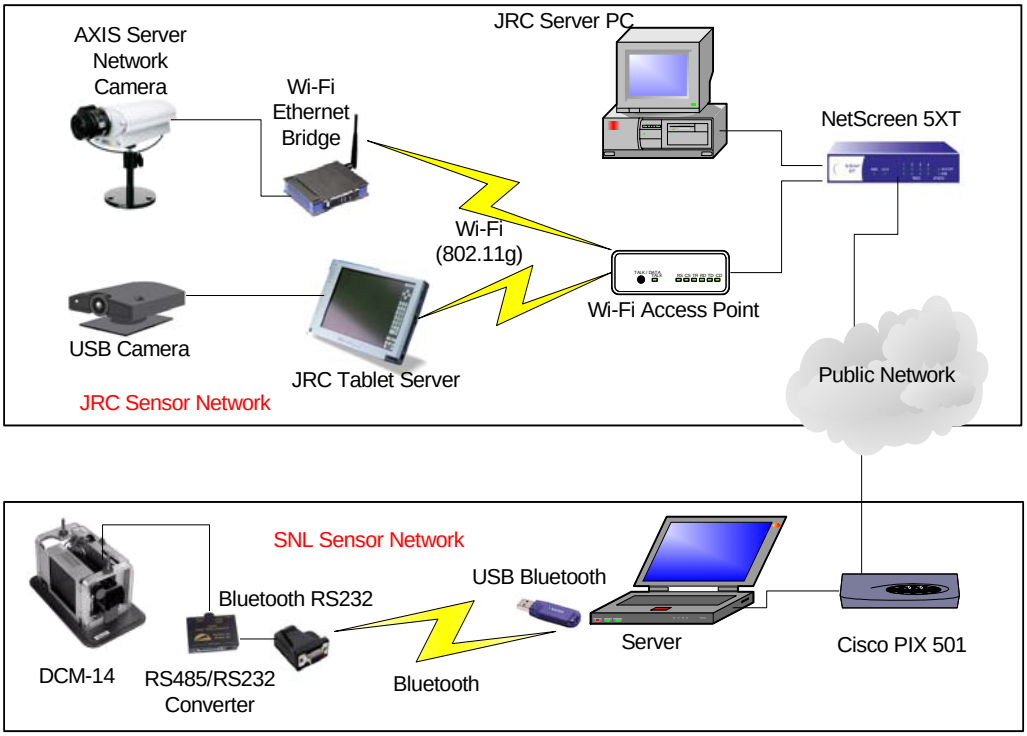
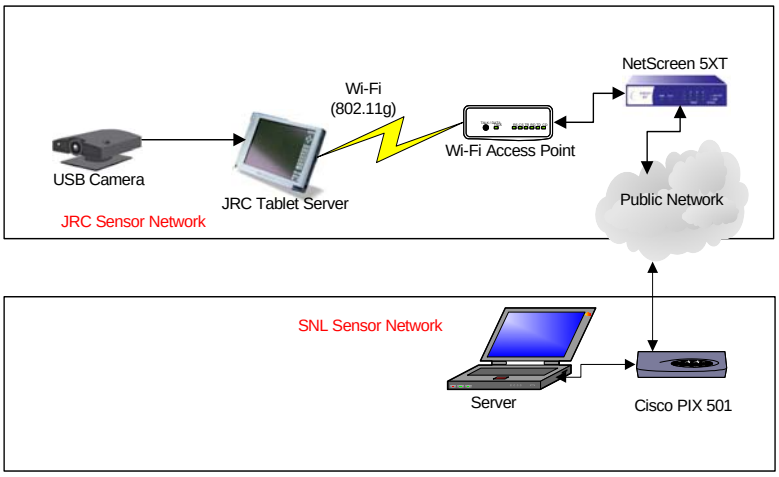
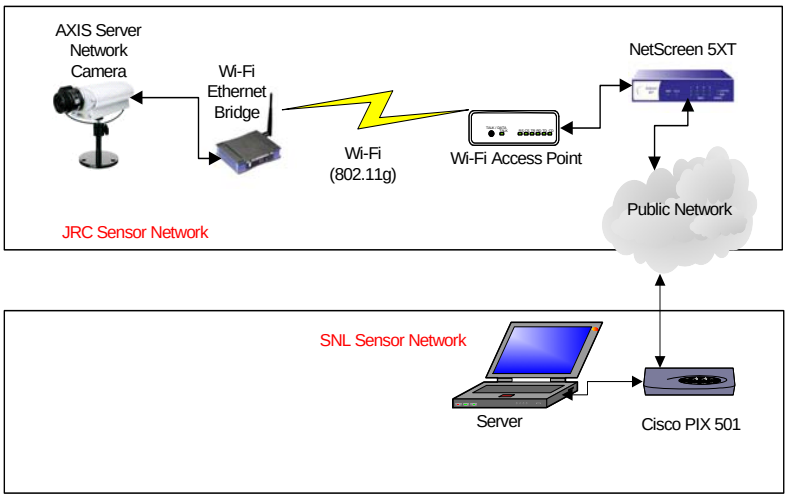


Figure 2: Architecture of SNL and JRC sensor networks.



(a)



(b)

Figure 3: a) Videoconference - b) Live Remote Surveillance. (See Figure 2 for larger print).

Experiment 4 – **Remote Surveillance** (Figure 4.b):
Users logged into the JRC Tablet Server or at the JRC Server PC connect with the secure VPN link between the two gateways to the SNL server, view and download the DCM-14 archived images (more than 5300 images for around 40Mbyte). This takes a few minutes.

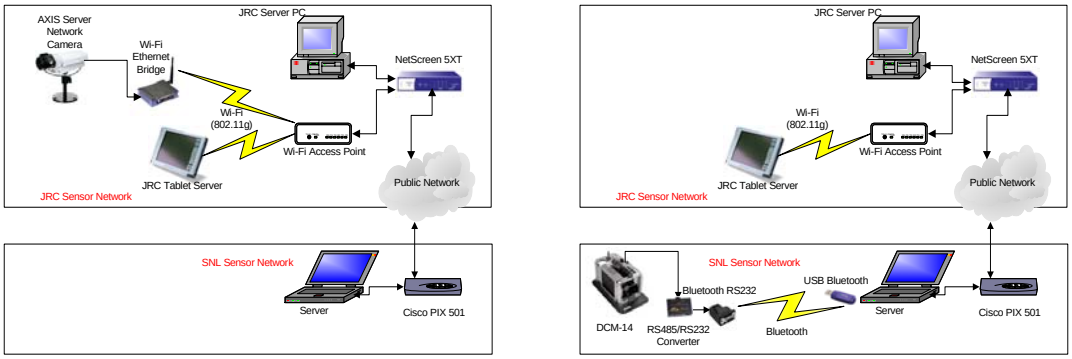


Figure 4: a) Future inspection - b) Remote Surveillance. (See Figure 2 for larger print).

All the data transmitted and received between the IPSec VPN gateways are encrypted and authenticated with the parameters described below.

IPSEC PARAMETERS:
In order to establish the VPN connection, the IPSec parameters to be used should be agreed upon. These parameters were configured using the NetScreen GUI, and are shown in Tables 1 and 2:

Parameter	**Value**
Encryption algorithm	**DES/3DES**
Diffie-Hellman group	GROUP 2
IKE mode of authentication	PRE-SHARED password
Lifetime	28800 SECONDS (8 HOURS)

Table 1: IKE parameters

Parameter	**Value**
Authentication and Confidentiality protocol	ESP
Encryption algorithm	**DES/3DES**
Hash Function	SHA (Secure Hash Algorithm)
Integrity algorithm	HMAC (Keyed-Hashing for Message Authentication)
Security Association Lifetime	3600 SECONDS (1 HOUR)
Perfect Forward Secrecy	**GROUP 2**

Table 2: IPSec parameters

These parameters adapt the VPN level of security to the desired requirements. A higher level of security normally implies a lower performance. As such, it is necessary to establish a compromise between both. In this way, 3DES was selected as the secure encryption algorithm, though it is more computationally intensive than DES, which is no longer considered a secure encryption algorithm.

Within the SNL sensor network, the Bluetooth devices were evaluated with respect to security using common methods (mentioned in references 2 and 6). The devices examined consisted of a Free2Move serial-to-Bluetooth adapter (DCM-14 side), and a D-link USB Bluetooth adapter on the SNL server. The D-link USB adapter could be configured to run in discoverable mode, non-discoverable mode (other devices must explicitly know a device's MAC address to enable a

connection), and a mode in which it requests connection to any discoverable Bluetooth device within range. However, the Free2Move device could only run in discoverable mode.

The initial security audit was done by simply walking within range of the sensor network with a Bluetooth enabled computer and querying for discoverable devices. This revealed the MAC address and the feature set of the Free2Move device since it could only run in discoverable mode. This audit did not reveal the D-link adapter attached to the server as it was configured to be non-discoverable. However, when the Bluetooth enabled computer was itself put into discoverable mode, the server attempted a connection to it also revealing its MAC address; the server was configured by default to attempt to connect to any discoverable device. Next, the D-link adapter was placed in a mode to not attempt to connect to other devices. From this change, the Bluetooth enabled computer was unable to gather the MAC address from the server. However, using a software package called Redfang [9], which runs on a Linux platform, the server device's MAC address was discovered using a brute force method.

With the knowledge of a device's MAC address, it is possible to pair to that device. This can be done by directly creating a connection using a raw Bluetooth connection command and bypassing the normal Bluetooth pairing sequence. There are sets of Linux tools from Bluez [10] that can enable this pairing with a set of straightforward commands. This was attempted with the Free2Move device, and it did indeed allow a pairing. However, when the features of the Free2Move were set to only allow pairing with one device, this connection could not be made. A test was also run to establish if a connection could be made to the D-link device, but the test failed. Had the pairing been allowed, this connection would allow downloading and uploading or any other Bluetooth feature supported by the D-link adapter to the server.

The overall security review showed that the available features of a device could directly determine the security of the system. The final configuration of the Free2Move system only allowed pairing with one other device, and once the devices were properly paired, illicit connections were not possible. So, if the Bluetooth device's features allow for the limited pairing to only a single device or a specific set of devices, it would not be possible to create an illicit pairing. Unfortunately, the majority of Bluetooth devices do not allow this limiting feature, as it is not a common part of the Bluetooth specification. Therefore, for devices without the limiting feature, additional methods of security must be used, such as VPNs or the use of encryption technology and authentication within the sensor devices themselves. Also, by configuring each device to the highest level of security possible, illicit connections can be more difficult.

Concerning reliability, the transmission of the DCM-14 images to the server was reliable. The DCM-14 was configured to take images at 15-minute intervals, thus sending 96 images per day to the server. For a period of over three weeks, all expected images were received. JRC also reported a very reliable network.

Neither JRC nor SNL performed formal interference testing. However, for SNL to informally gauge the susceptibility to interference the DCM Set software was opened, and the "movie" command was run (poll DCM-14 and send a live CCD camera image approximately every 5 seconds). This means that the Bluetooth devices were nearly constantly active. Another laptop with a Bluetooth device connected to the Internet through a Bluetooth-enabled cell phone, and the user surfed the web, continuously sending data. No interference problems were noticed. Another interference test was to again run the movie, and use a Bluetooth cordless phone at the same time to see if the quality of the call was reduced, or if the server had trouble collecting the image. The voice quality remained audibly unchanged, and the images continued to transfer. If interference had been noticed, the previous mentioned mitigation methods could be used.

4. Ongoing and future work

JRC has recently installed an Oracle server and ESRI software (Geographic Information System, or GIS, software) to build a geo-referenced database. Some additional sensors and GPS have been installed as well. JRC is currently entering geographic data into the database. The concept is that the

JRC Tablet Server users can move around with some sensors (e.g. GPS, laser distance meter, camera, etc.) and transmit/receive data via the IPSec link from/to the SNL VPN side and/or the JRC PC server. This data, as well as the stationary sensor data, will be collected into the database and published onto a VPN-protected GIS website that only authenticated users can access.

In a future experiment (Figure 5) a 'sniffer-machine' connected to a hub just ahead of the JRC VPN network will be used to verify the security of the data transmission between JRC and the SNL VPNs.

In addition to the experiments performed between SNL and JRC, the IAEA has expressed interest in establishing a demonstration wireless sensor network in Vienna that would join the established secure SNL/JRC network. The components proposed by SNL and JRC to include the IAEA are, (1) a PC with a VPN at the IAEA that would be connected to both the SNL and JRC sensor networks, (2) sniffer software would be used to evaluate security to build confidence, (3) a user with a tablet PC at the IAEA would access the SNL/JRC sensor networks using either Wi-Fi or GPRS (mobile data network) to gain access to the secure network (this allows user mobility), (4) build an IAEA wireless sensor network, and (5) setup a Mobile Information System for Safeguards Inspectors.

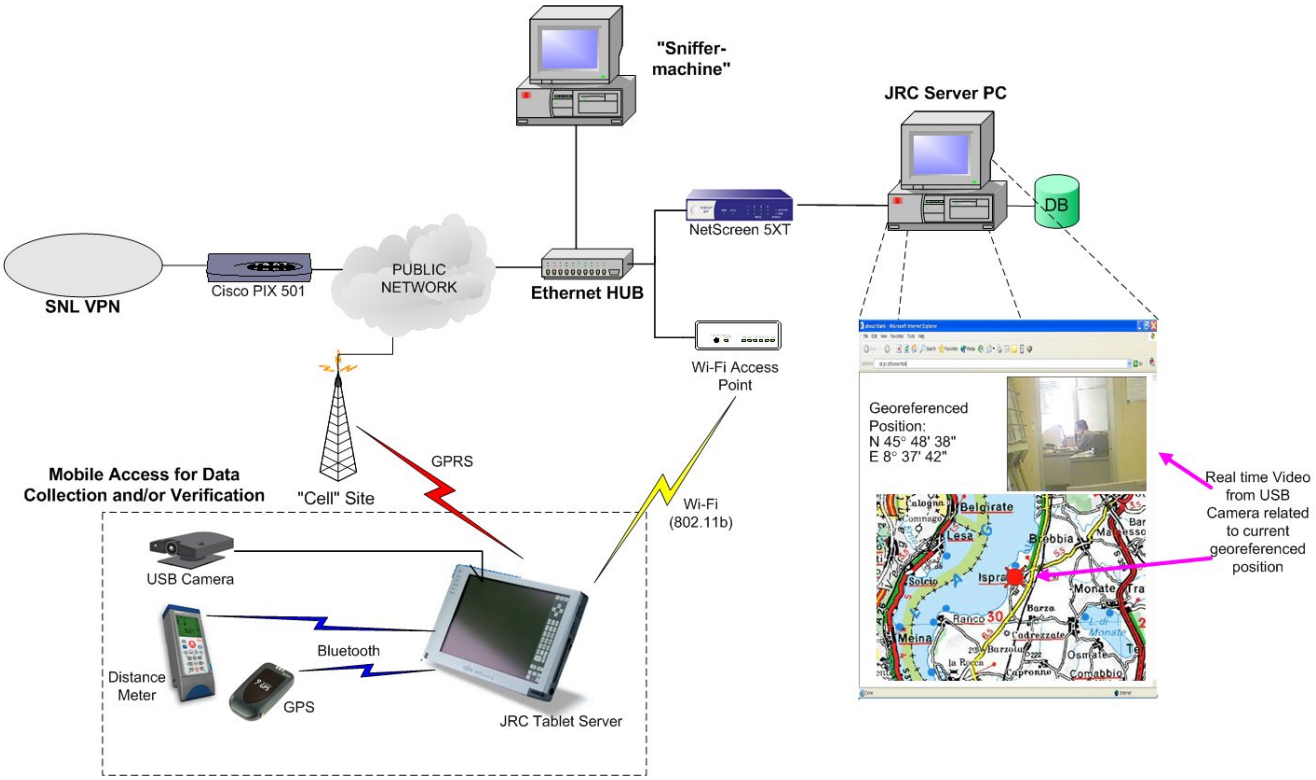


Figure 5: Architecture for ongoing and future experiments.

5. Conclusions

A major cost factor in the installation of monitoring facilities is the cabling necessary to tie the network together. In many cases, this installation is extremely intrusive to the facility operator and can even

create Environmental, Safety and Health (ES&H) concerns. Furthermore, cabling can be a major problem in follow-on inspections, as it is often run in the least accessible locations.

Designing and installing a safeguards system that relies heavily on wireless technologies offers many potential benefits. These include: reduced installation costs, reduced maintenance costs, reduced intrusiveness for plant operators, and easier system inspections.

The use of wireless technology, however, has been resisted due to two major concerns. These are the potential interference with other plant equipment operations, and the potential loss of security of the transmitted data. To evaluate the issues involved with wireless technology, SNL has collaborated with JRC in a joint technical field trial. Informally, interference was not a problem, and methods have been listed for mitigation. Both SNL and JRC's sensor networks and VPN networks were reliable. Finally, with proper configuration and security devices, the network was secured.

Based on results to date, wireless networks show the promise of providing improved safeguards effectiveness, while simultaneously reducing networking costs within a nuclear facility.

6. References

1. http://www.sss-mag.com/bluetooth.html
2. Jakobsson, M., Wetzel, S., *Security Weaknesses in Bluetooth*, Lucent Technologies – Bell Labs.
3. http://computer.howstuffworks.com/wireless-network.htm
4. http://www.wi-fi.org/
5. http://www.wi-fiplanet.com/
6. Xydis, T.G., Blake-Wilson, S., *Security Comparison: Bluetooth Communications vs. 802.11*, Bluetooth Security Experts Group, 2002.
7. Hodgdon, C., *Adaptive Frequency Hopping for Reduced Interference between Bluetooth and Wireless LAN*, White Paper, May 2003.
8. http://computer.howstuffworks.com/wimax.htm
9. http://www.atstake.com/research/tools/info_gathering/
10. http://www.bluez.org/

Castor® containers: Ultrasonic sealing system for dry applications

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Abstract:

New ultrasonic reading heads have been designed at SILab. The heads include a new motor drive with an encoder for precise feedback on the reading positions. The software and acquisition data system were developed as well. The heads are suitable both for dry and underwater applications. The head for dry reading includes a water reservoir to allow the coupling with the ultrasonic bolt. Also the weight and dimensions of the heads were reduced to ease transport and handling. One prototype was fabricated at SILab for each system.

On request of DG-TREN, a new ultrasonic sealing system for dry applications is under development. It is based on the characteristics of the CASTOR® containers (dimensions, ambient and condition of operations). It includes a new upgraded ultrasonic sealing bolt, the reading head and a remote positioning system to reduce to a minimum the dose to the inspectors. The seal itself incorporates several innovative elements to avoid tampering with. A full scale mock up is under fabrication.

Keywords: Ultrasonic seals; Underwater applications; Dry applications; Castor® containers; Remote control;

1. Introduction

Safe and effective storage of nuclear materials is one of the most important issues in nuclear safeguards. Checking the location and content of each nuclear fuel container must be done regularly. The Seals & Identification Laboratory of the Joint Research Centre of the European Commission (*SILab*) has developed technologies based on ultrasonic and transponders suitable for this scope.

SILab has already developed ultrasonic seals and equipments for underwater applications that are in routine use by both agencies (IAEA and DG-TREN/ESO) in Sellafield (UK) and La Hague (F) installations.

These equipments have been installed several years ago, and a review is undergoing. A first step has consisted in the substitution of the reading station. New reading-heads and software have been developed.

On request by DG-TREN, a study of the application of ultrasonic seals in dry conditions has started. Ultrasonic seals present as main advantage their stability against time and radiations. Being purely mechanics and from the same material than the container casks they are supposed to withstand the life time of the containers. Only the reading equipment has to be maintained. On dry applications, where the seals are supposed to remains for a long period of time (more than 10 years), and where cumulative radiations can be high, they could present great advantages. The drawback is their accessibility that makes them easier to be attacked.

2. Equipments for underwater applications

SILab seal reading system has three main components: the ultrasonic bolt, the reading head and the data processing and acquisition system. All these components have been critically reviewed and simplified to ease their use and to reduce costs.

New seals have already been developed and tested. After two year tests in laboratory, they have been supplied to DG-TREN inspectorate and are in routine use in La Hague since March 2004.

2.1. New reading station

The configuration of the system has been reviewed. The use of the ultrasonic board directly coupled to an industrial computer is very compact but presents the drawback of high cost and market availability.

The computer market is in constant evolution, only some specific models of the above brands are suitable for our purposes. If in the future these computers will be discontinued we will have to face a redesign of the system.

The system used in the past has been transferred to a laptop-based architecture using a bridge between the existing ISA board and a laptop-compatible bus (e.g. PCMCIA). In this scenario no changes are necessary to the software part of the system, and the existing controlling application can be reused.

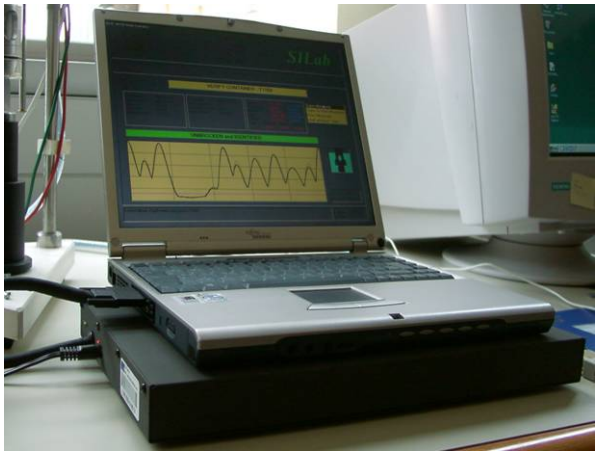

Photo 1: New reading station

Below the portable computer the interface box containing the ultrasonic board is clearly visible. This solution solves the cost aspects of the industrial computer and also the power supply one. New computers have less and less power available. Such interface box can be built with the required power supply.

This system has been fully tested at *SILab* and is now routinely used by DG-TREN since March 2004.

2.2. New Reading-Heads

The new reading-heads are built around a new smaller motor with encoder and a compact gear transmission that increases the performances of the head while decreasing the overall dimensions. The ultrasonic transducer has not been changed. The head is more compact and lighter.

After having tested the mechanics of the new reading head, several prototypes were manufactured to choose the best compromise between mechanical arrangement of the drive and electronic control system. Care has also been taken to make the new reading heads compatible with the old one: Old and new heads has to be compatible and shall give the same reading on bolts. Old bolts read with new heads shall be still recognizable.

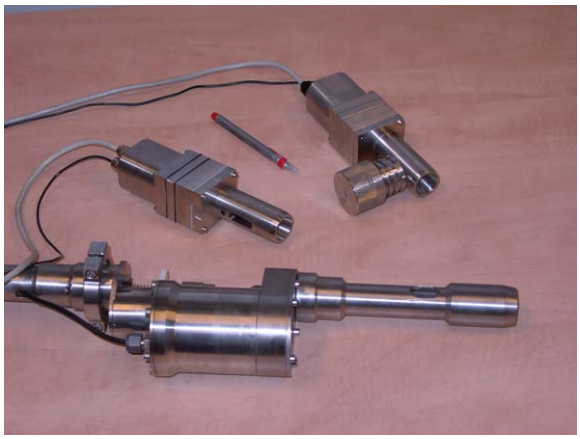

Photo 2: New (and old) Reading-Heads

The photo shows an old Reading-Head (front) with two new ones. One of the new Reading-Head (upper-right) is modified to operate in dry condition. The head for dry applications has a water reservoir that allows the coupling between the ultrasonic transducer and the bolt head. The reservoir, in the present configuration, allows about 20 readings before being recharged.

The advantage of these new heads is the contained overall dimensions and the light weight that make them more attractive for transportation. The new Reading-Heads are compatible with the new reading stations described before.

The test campaign of these new reading-heads gave satisfactory results.

2.3. New software

The inspection software has been adapted for Windows. It has the same functionalities as the actual inspection software that run under DOS. It is already adapted for the new reading-heads.

This new software is compatible with the new and the old bolt designs. The software automatically adjusts its parameters to the two bolt configurations in a transparent way for the inspectors.

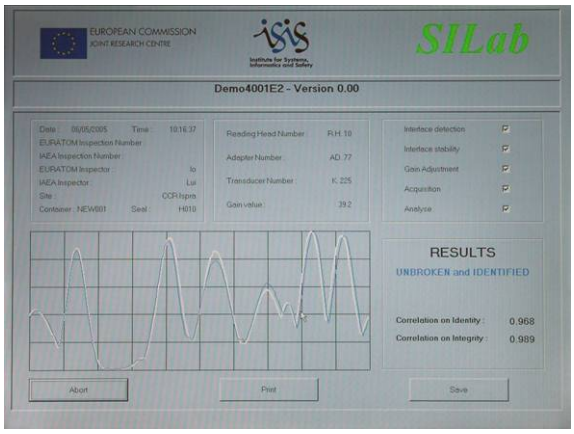

Photo 3: Inspection screen of the new software

The software is written for Windows® 95/98. It has been tested for already one year. A version for Windows® XP is under development.

3. New sealing system for dry application with increased security features

A new seal, with increased security featured, suitable for dry applications on CASTOR® containers, has been developed at *SILab*.

This development has been commissioned by DG-TREN. The scope is to develop a seal for containers that can stay for years (a minimum of 10 years). Such seal should be used in dry storage (either permanent or temporary), in a highly radioactive environment. It must be as low maintenance as possible and should be easy to operate by inspectors and operators. As guidelines for the development, the characteristics of the CASTOR® containers (mechanical geometry and characteristics, and use environment) have been taken. The seal must be placed on the primary lid in place of a standard bolt without any modification on the cask body or on the lid. He must be verifiable remotely.

3.1. The seal

The sealing bolt for CASTOR® storage uses an identity core, including a rupture element used as integrity indicator. This core is embedded into an ultrasonic bolt with the following innovative features:
- The core is mainly housed below the level of the upper surface of the cask lid.
- The seal head is provided with a second identity, correlated univocally to the analogical one in the core. This second identity consists of a number of holes drilled inside the head at different depth and pitch, to give a univocal ultrasonic response.
- The integrity element is broken applying to the seal head either a torsion torque (normal unscrewing of the sealing bolt) either a traction force.
- The seal body is almost completely hidden in the lid
- The seal head contains a RF transponder allowing saving all the data of the installation for a limited period of time (typically 1 years):

The sealing bolt should have a mechanical strength similar to the other bolts that close the CASTOR® upper flange.

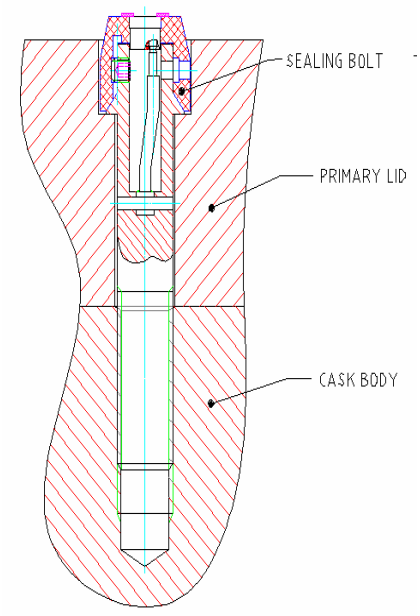

Photo 4: Dry Castor® ultrasonic sealing bolt.

During the installation, a special tool tightens the bolt with the required torque. The torque and all the data relevant to the installation are stored into the transponder of the seal. This allows the installation to be done by the operator and the inspector presence is not strictly required. The inspectors will retrieve and control these data during their first inspection.

After installation, normal inspections consist in reading the two identities using the ultrasonic reading head. The two identities of the seal are correlated together univocally.

Photo 5: The two identities of the seal

The seal is yet under qualification and commissioning and it will be ready for testing soon.

3.2. The Reading-Head:

This new Reading-Head is specifically designed for the reading of the two different integrity features integrated in the new dry seal.

The design enables the reading of the two integrities located on two different radiuses using only one ultrasonic transducer. The transducer is moved from one radius to the other by a cam assembly when the main actuator changes direction of rotation.

The ultrasonic part of the Reading-Head is identical to the one used on the underwater systems that are already in routine use. The motorisation part is also equivalent to the one used in the new Reading-Heads that are used for some years in *SILab*. The innovation is the device used to change the radius of the readings with the change of rotation direction.

Photo 6: Assembly of the test Reading-Head

A test version of the reading-head concept has been built and is under qualification. The definitive reading-head will be based on the same concepts used for the tests.

3.3. The software

The software is based on the new software for underwater applications.

The two fingerprint curves are correlated and overlapped. As shown in the next photo, any change in one of the curves or any change in relative position will dramatically change the bolt response triggering an alarm. The white curve is the analogical fingerprint and the blue one is the seal head fingerprint.

At any attempt of removing the sealing bolt, the integrity will break and the software will detect the opening attempt

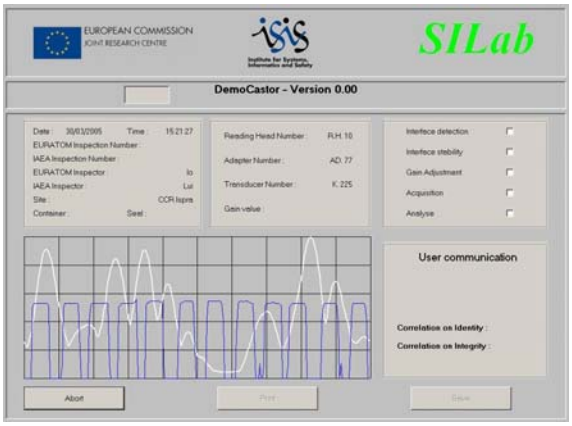

Photo 7: Reading screen of the Castor® software

This software is written for Windows® 95/98. A Windows® XP version is under development.

3.4. Remote positioning system

Positioning manually the ultrasonic reading head to read a container seal could be a task either difficult or time consuming. During all the reading operations, the inspector and the operator support team will be exposed to radiations.

An automated and remotely operated positioning system, guided to the target through a CCD camera and a force/torque sensor, has been developed at *SILab* (see photo 8).This positioning system is able to recognize, with the CCD camera, the seal in an area of one square meter and to automatically drive the reading head to couple the seal head. The positioning system is simple and light. It is suitable to be hanged directly on the bridge frame where the operator was used to carry out the manual operation.

Photo 8: General view of the remote positioning system in *SILab* laboratory

With this equipment, it is sufficient to reach with the bridge a position relatively close to the bolt ($1m^2$) to be verified, the vision system will automatically recognize the sealing bolt and will drive the Reading-Head close to the bolt (inside $1cm^2$ of precision). The force sensor will control the final approach and will stop the movement when the final position is reached. Once the reading-head is put on the seal, the inspection software will perform the reading.

The system may also be interfaced with the operator crane control. Once the operator crane has brought the positioning system roughly upper the container, it can send a signal to its control that will perform the approach and the reading. Once the positioning system is back in safe position it will send a corresponding signal to the operator control system in order to go to the next container.

The positioning system is designed to be maintenance free. The design can be optimized, if required, to satisfy special safety requirements, space or installation constraints.

A demonstration of this remote positioning system is operational at *SILab* since 2004.

4. Next steps

The mechanical tests are undergoing. They are aimed to test that the seal, being also a bolt,

can resist to the "natural" constraint put on the bolts (torque and axial force).

Tampering with tests are also undergoing. The scope of these tests is to try to tamper with the seals and to figure out how these tampering can be detected. Depending on the results of these tests some modification of the design of the seals may be necessary.

Ambient tests will also be performed. They concern the seals themselves and also their relative's equipments. These tests will be performed at JRC/Tempest® laboratory, which is fully equipped for thermal, electromagnetic and mechanical tests.

A vulnerability assessment of the seals and relative equipment will be conduct independently by the IAEA to determine the vulnerabilities of the current design and application.

A new ultrasonic board that can be interfaced with the computer using the USB bus should be available soon. This board will allow building more compact electronic control using external power supply. Such control will also be easier to handle and to use.

Designing a Special Monitoring Systems for an Unique Application (Chernobyl Shelter)

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Abstract

The IAEA has the responsibility of establishing safeguards on the Unit 4 reactor buildings during and after the preparation for and the build of new shelter for the unit. However, the problems presented by this project are far different from "normal", "standard" safeguards projects. As a result, the IAEA has declared this a "special" facility, which means that innovative means must be developed to effectively and efficiently provide safeguards for this area.

The significant safeguards focus at this facility is a significant number of irradiated fuel assemblies left in the reactor hall. As work on the stabilization of the existing structure begins and as the new shelter takes shape, more workers are going to continually access this area. This activity provides a challenge to safeguards because of the increased number of people and the increased activity in this area.

The inspectorate is best able to define an innovative approach if they have creative support from the instrumentation support people. This support includes novel adaptations of existing IAEA inspection equipment as well as support to characterize the actual radiation fields in the facility so that the adapted instruments can properly couple with the radiations signatures to allow definitive, decisive and defendable safeguards conclusions to be drawn about the facilities compliance with safeguards responsibilities.

This paper will briefly review the approach being taken to safeguarding this facility, and then go on to describe preliminary tests and measurements that were done to set the foundation for the specification of the monitoring system for this facility, and finally the integrated video and radiation based system and its implementation will be discussed.

Keywords: Unattended Monitoring Systems, Chernobyl Shelter.

Introduction

Before workers started shoring up the existing structure of the Unit 4 reactor at Chernobyl NPP, the access to materials of safeguards significance at this facility was severely limited. As the strengthening of the reactor building progresses in preparation for the building of a Shelter that will be used to shelter and contain the materials as the crippled structure is removed; one can envision credible scenarios for the diversion of an accountable amount of special nuclear materials. Thus, the IAEA is looking at a four step approach to safeguarding this material. The first step is to monitor the access points for the contractors and equipment removal from the Unit 4. There are two personnel access points and a single equipment access point. The IAEA is installing monitoring equipment at all three points to assure that no nuclear material is inadvertently or purposely moved out of unit 4 through these portals. A second step is to work with the facility to identify nuclear materials that are located and identified during excavation

activities done in preparation for the building of the shelter. Identified radioactive will be classified as nuclear or non-nuclear and separated into safeguarded and non-safeguarded groups. The third step is to enhance the safeguards to include a means to verify that the spent fuel remains where it is during the time of the strengthening of the structure. During this time, cranes that are capable of moving the overburden to expose the "buried" fuel assemblies will be introduced into the area, and hence the IAEA must provide a means to assure that the spent fuel assemblies are not moved. The final step is to provide safeguards during the de-construction of the reactor hall and structure after the shelter is in place. The present paper will address the monitoring system planned for step 1 and the preliminary measurements carried out to define the monitoring concept for step 3.

Shelter Access Points

Although construction is already on-going at the site, the IAEA was able to hand carry instrumentation to each access point and determine the amount of background neutron and gamma radiation. The fields of view of cameras that would be installed in conjunction with the radiation instrumentation were also assessed. One side of a standard neutron coincidence collar that is normally used for fresh-fuel measurements was carried along with a MiniGRAND instrument to acquire neutron data in the areas where monitoring instruments are proposed to be installed. In addition a hand-held HM-5 gamma-ray instrument was carried to investigate the background in these areas. The instruments detected only low background neutron and gamma rates that will allow IAEA detectors to operate at the access points without significant shielding requirements and that will allow the system to have adequate detector sensitivity.

Instrumentation Overview

The monitoring system to be implemented at these points is shown in figure A. The figure shows an inspector's room in the upper left which has the central collection computer that is the interface between the inspector and the distributed autonomous sensors located at the access points. The conduits that link the cameras, instruments and

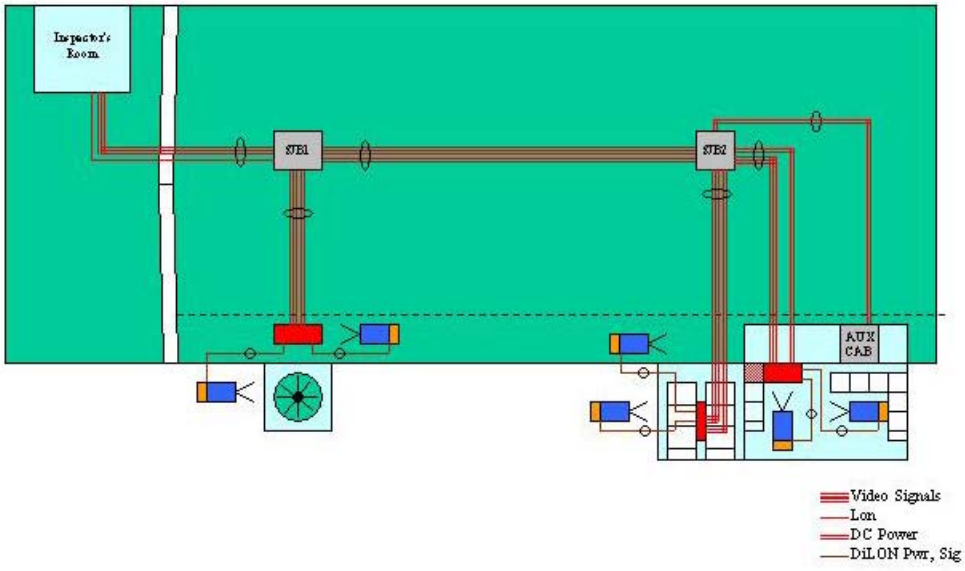


Figure A. Shelter Access Monitoring Points.

Collect computer are to physically protect the cables. The cables do not need to be secure because video data frames are authenticated at the camera, and the communication among the radiation instruments, the cameras (trigger signals) and the collect computer are authenticated.

The first access point on the lower left is for personnel only. It has a badge reader control so the people have to pause for a short time at this point. On the lower right area are two access points. The larger room on the right is a freight elevator, which is really a jib crane. Just to the left of that is another personnel access point, but it is for emergency use only. At each of these points an integrated detector enclosure, including detectors, supporting electronics and batteries, will be installed. The integrated detector is an autonomous unit that can collect data for a limited time (less than one full inspection period) even if it is completely cut off from the rest of the world. These detector enclosures are shown in Red. The camera enclosures are also autonomous, and at each measurement point, two cameras are used to provide video coverage that will supplement and clarify the radiation signatures. While they are used primarily for support, they are totally independent; hence the frames they generate could be reviewed with the standard video surveillance software if required.

Video Package

As was mentioned, the sensor complement is identical in all three areas. This includes two standard video electronics packages each with a special camera with built-in infrared illumination. The same type camera has been successfully used and demonstrated with the MMCT, the rail transport car which is used to move spent fuel among the different spent-fuel storage areas at Chernobyl. Another implementation of standard equipment in the safeguards configuration is the inclusion of a gelled-electrolyte battery and charger system inside an IAEA standard camera housing. This battery-charger system was originally used with the standard MiniGRAND instruments. This battery-charger system powers the 12-volt camera/infrared illuminator, the DCM-14 camera support module and an instrument network node called the ILON, for 4 – 5 times the amount of time the standard DCM-14 Li-Ion battery would last when used in an integrated monitoring system. The functional capabilities of the DiLON camera enclosure are shown in Figure B.

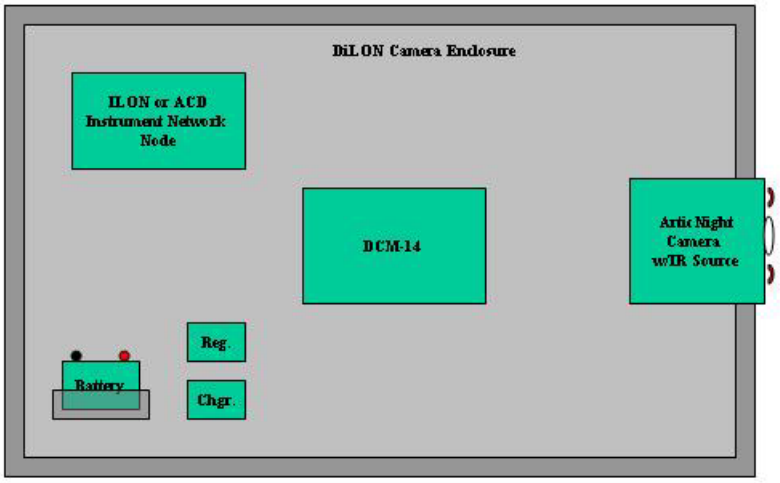


Figure B, the Functional DiLON

The Camera housing has all the essentials to allow video information to be collected, saved and protected even though all power and other signals are severed from the camera enclosure. Such features minimize the likelihood the IAEA and the facility would have to provide manpower for an inventory re-verification in the event power was cut by accident.

The camera enclosures are connected to a central data collection computer though electronics that is standard usage for the DMOS camera systems at the IAEA. The software complement that runs on the collect computer is taken from the standard DMOS software. There are some data architectural differences in which the data are retrieved, saved and offloaded, but all the basic video data collection software is unchanged.

Radiation Package

The IAEA has for almost two decades exclusively used gross-gamma and gross-neutron radiation signatures for unattended-monitoring purposes. For the special situation presented by this facility and with recent developments in NaI stabilization it seemed reasonable to apply this technology in lieu of the normally used, extremely robust ionization chambers that would not be sufficiently sensitive for looking at low-dose gamma signals. The integrated detector enclosure functionality is shown in figure C. This enclosure contains a neutron slab detector with six 1-meter long He-3 tubes, a 10cm cubic NaI detector and MiniADC monitoring instrument with a battery and charger circuit.

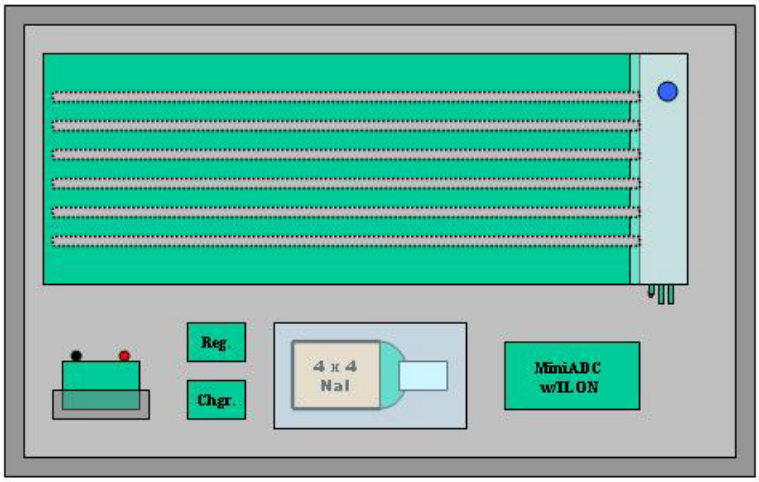


Figure C. Integrated detector enclosure

In normal safeguards, the sources that pass a sensor are generally of a specific type and well characterized, such as spent-fuel, fresh-fuel, and normally the fuel with be either LEU or MOX. Any irradiated non-fuel would typically also be well characterized in size and shape. At this facility the fuel possibilities are the same, but there are going to be pieces large and small, and there will also be irradiated or contaminated non-fuel materials. For well characterized fuel from a normal facility where there are regular cycles and a well known history, one can normally distinguish among the possible sources that pass a detector with gross gamma and neutron detectors. This is sufficient to allow the IAEA to verify the operator's declaration and to assure that no other activities that might be an indication of an attempt to divert material have occurred. At this facility, very little material that will be seen will be normal in the sense as described above. Therefore, it was deemed reasonable to upgrade the gamma sensors from gross gamma to

spectral-based detectors. The advantage of spectrum collection is that separate regions of interest can be identified. Different threshold can then be assigned gamma-energy regions.

Implementation and testing

The IAEA has built a test prototype detector system of the type to be used at the access ports to test its sensitivity and robustness and to develop setup procedures to determine the optimal configuration of the instruments. The system sits along an automobile access portal at the IAEA headquarters in Vienna. The test system has a small computer (that will not be used in the installed detectors systems) built into the detector enclosure to act as a collect computer for both the video and the neutron and gamma-ray data. The data are then transmitted to our laboratory at Headquarters using an RF Ethernet link. This test setup is shown in Figure D.

Figure D. The Chernobyl Shelter Access test detector system

Unit 4 Reactor Hall

The reactor hall in unit 4 contains the material that the IAEA has specific interest in safeguarding. This is the intact spent-fuel assemblies that are in the spent fuel pond buried under a small mountain of rubble. After having discussed the possibilities, it was decided to try to introduce video surveillance and radiation monitors. There are many problems with implementing either of these approaches. It has been decided that monitoring can be installed at the foreseen penetrations in the reactor hall wall between units 3 and 4. This wall is about 26m away from the spent fuel site, not considering the change in elevation, only the distance in the x-y plane.

Add to this situation the difficulties that there is no dependable lighting system in the area, and top it off with the fact that there is a dust suppression system that routinely puts a fine mist of water mixed with a conformal coating material that will play havoc with a lens/window system, and one ends up with quite a safeguards challenge.

The fuel to be safeguarded is one cycle burned fuel and it has been cooling for almost two decades. Due to the distance between the detectors and the material and the significant background in the area, the capability to detect any material movement is questionable. As mentioned before, spectral gamma capabilities are possible, spectrum characterization is still not fully clear.

The focus of the proof-of-principle efforts is to verify whether it is possible to visually view the area around and above the spent-fuel pond using illumination and camera positions located more than 26m from the spent fuel pond. However, another problem exists for using the cameras in the penetrations, namely neutron radiation that may affect electronics. In the 50mm penetration a significant neutron background was detected. Further more, this neutron flux peaks inside the wall. So shielding may present a significant problem given that the diameter of the penetration is limited.

He-3 detectors can be effectively shielded from the flux in the wall because the neutron flux within the wall is probably already thermal in energy. An “axial” detector where the tubes lie along the axis of sight is presently planned. This is shown in Figure F.

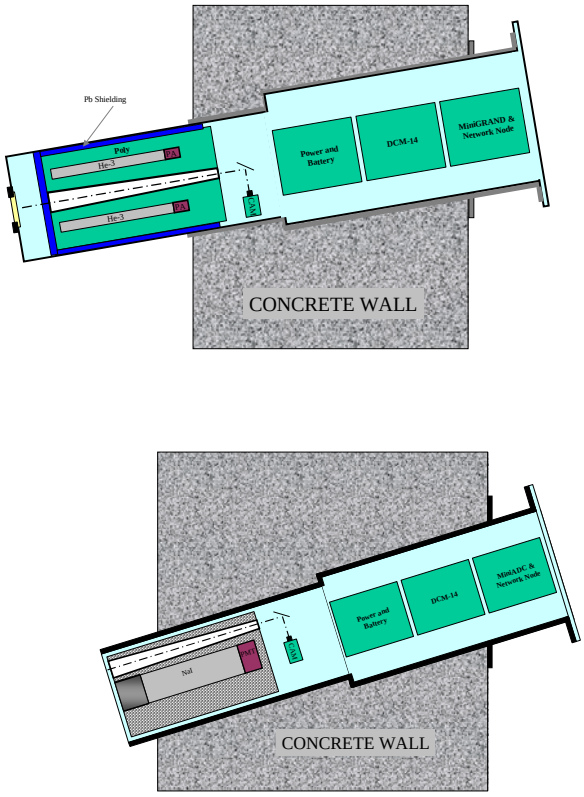
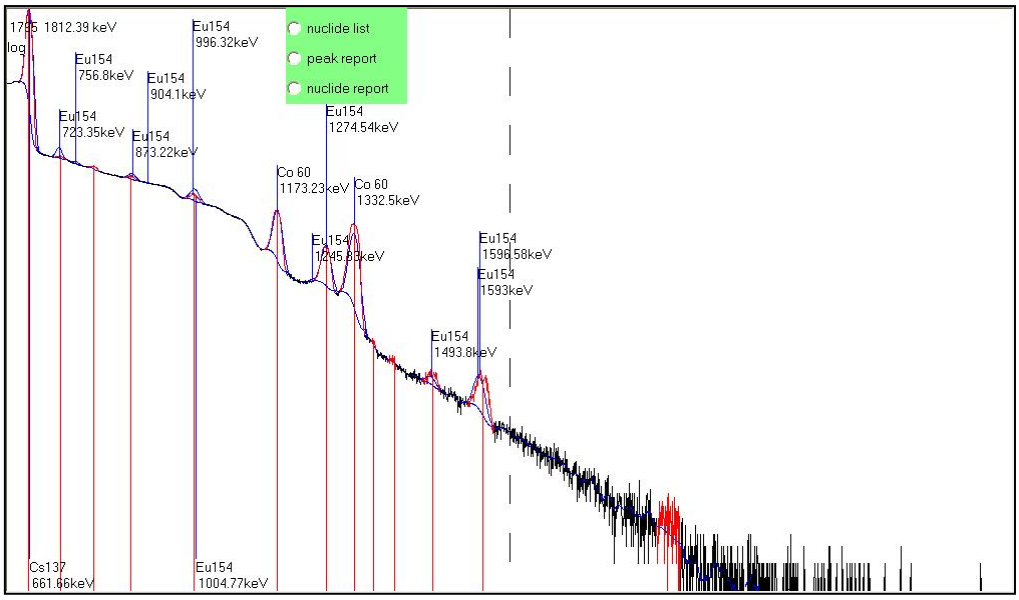


Figure F. The conceptual designs of a potential combined radiation and video sensor penetration.

Along with or in place of the neutron detectors IAEA is considering placing NaI detectors in penetrations to peer into the darkened reactor hall. By using columinators that view just over the top of the spent fuel location, it may be possible to detect the lifting of any of the materials by a significant change in the monitoring data. To investigate the real feasibility of this, a prototype detector will be built to make measurements through the wall. The columinators and possibly the detector will be placed though and inside the 300mm penetration.

Figure G: Sum spectrum from reactor hall measurements



Conclusions

The monitoring system for the Chernobyl Shelter project provides a very good illustration of how standard IAEA components can be integrated into novel adaptations to meet innovative approaches.

An Unattended Approach for the Safeguards of Spent Fuel Transfer to Interim Dry Storage in CANDU reactors.

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Abstract:

The transfer of spent fuel to dry storage canisters in CANDU facilities requires a substantial effort to IAEA Inspectorate, due to the requirement for inspectors on site to follow all the steps of the process. An alternate approach based on unattended equipment has been developed and field-tested during the last transfer campaign in Cernavoda Nuclear Power Station. The equipment utilized is a combination of surveillance and NDA systems, performing attribute testing and item counting. The transfer process can be subdivided into three major sub-processes: i) loading of the transfer basket in the spent fuel bay, ii) drying, welding, and transfer of the basket to the dry storage facility, iii) lowering of the basket into the concrete canister. Specific radiation monitoring systems have been designed for each of these steps, including an underwater directional monitor used for verifying the fuel bundles during the basket loading, a very low power neutron counter/logger installed on the transfer flask, and a gross gamma counter system used to monitor the loading of the basket into the canister. Surveillance cameras integrate the NDA data collected and provide backup in case of failure of one of the radiation monitors. This paper will describe the technical details of the systems designed, review the results obtained during the transfer campaign, and discuss the future implementation.

Keywords: Unattended Monitoring Systems, CANDU Reactors, Dry Storage Transfer

1. Introduction

The transfer activities of cooled spent fuel from reactor wet storages to interim dry storage facilities represent a significant new operational issue for safeguards inspections conducted by the International Atomic Energy Agency. The current approach under a traditional SG regime requires the constant presence of inspectors to perform random verification of the irradiated fuel before encapsulation into the transfer container and to maintain the Continuity of Knowledge (CoK) during the transfer. This approach takes a lot of resources in terms of inspection days (PDI) and increased burden to the operator. Current estimates amount to 1000 PDI per year in 2005 just to safeguard transfer activities performed in OLR facilities.

An unattended approach based on installed equipment has been developed and field-tested to address the monitoring of the transfer in CANDU single-unit facilities. In these facilities, the fuel bundles are stored in the dry storage in cylindrical containers (baskets). A basket accommodates 60 SF bundles that are loaded at the underwater loading station in a specific area of the Spent Fuel Bay. The basket is then welded and transferred to the dry storage facility in a shielded flask mounted on a truck. A full basket has a total content of about 0.5 SQ of Pu. At the dry storage facility, the basket is transferred from the shielded flask to a concrete silo that can contain 9 or 10 baskets, depending on the configuration. Once full, a silo is welded and kept under dual C/S.

The purpose of the system that will be described in the next paragraphs is to verify the fuel before the transfer to a difficult-to-access location and maintain the CoK during the complete process. The design of the unattended approach has been developed with focus on the following requirements:

- limited impact on operator's activity during the transfer process
- high reliability
- full redundancy in case of equipment failure
- modularity to adapt the system to different processes
- provision for remote monitoring, where applicable

The total cost of the unattended system is around 150,000 US$.

2. Architecture of the unattended approach

Figure 1 provides an overview of the transfer process: for simplicity, the process can be separated into 3 subsequent steps. The first one takes place underwater in the Spent Fuel Bay and consists of the loading of a basket with 60 SF bundles. The second step includes the operations to remove the loaded basket from the bay, dry and weld it, and transfer it to the dry storage facility inside a shielded flask. The final step consists in lowering the basket from the transfer flask into the concrete silo. The various systems employed are indicated in the figure. The process takes a minimum of 3 hours to complete. It is usual to have interruptions and is common that a welded basket stays in the welding station overnight. It is also usual operational practice to have a partial overlap of the different steps: while a basket is moved to the dry storage, a second one can be filled in the SFB. Note anyway that no basket can be lifted to the welding station if the transfer flask is not positioned on top of it, being the transfer flask an integral part of the welding station shielding.

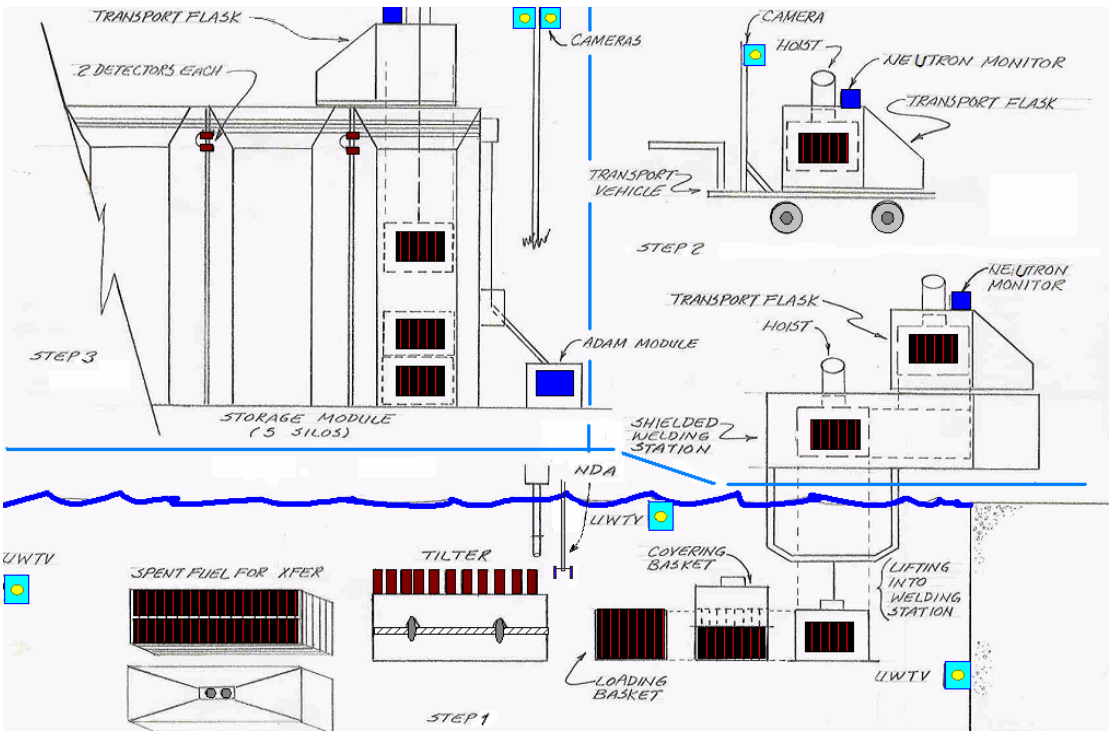

Figure 1 – Scheme of the transfer process.

2.1. Step 1 – Loading of a basket

According to Agency criteria, irradiated fuel designated to be transferred to difficult-to-access storage need verification with non-destructive attribute analysis (NDA) prior to loading of the containers. This requirement does not apply if fuel is kept under dual C/S systems in the wet storage and was successfully verified when first moved into the area by a Bundle Counter system. Unfortunately, dual C/S is not implemented in most of the CANDU single-station facilities due to operational reasons: hence attribute re-verification at the moment of basket loading is required.

In the new unattended approach here described, NDA verification is performed by using a pair of gamma detectors installed underwater in proximity of the loading station and connected to a VIFM timer-counter system. The two detectors are spaced about 50 cm in the direction of the fuel path so that the direction of movement can be detected. The operator moves manually the bundles by using a tool that is suspended to a fixed rail: the path followed by the bundles during the movement is therefore repetitive, although timing and speed can vary. Detectors are located about 80 cm from the bundle path. Figure 2. shows a typical signal pattern obtained while loading 12 bundles into a basket. Despite the increasing background produced by the basket being filled, it is possible to count the 12 bundles and verify the direction of movement. This preliminary result is expected to improve after the installation recently performed of improved collimators expected to reduce the background from both the basket and the tilting table.

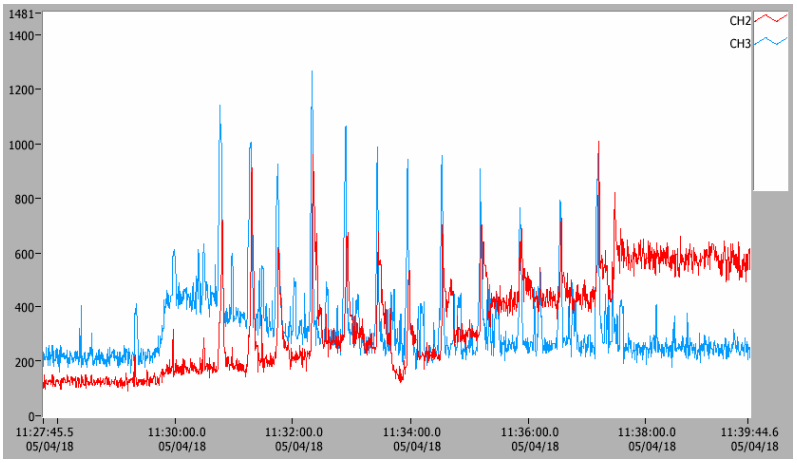

Figure 2. – Preliminary results from the NDA system in the SFB

The NDA system is complemented by two cameras mounted in semi-submergible enclosure and viewing the loading station from the two different sides of the Spent Fuel Bay. The purpose of the cameras is to cover possible diversion scenarios including removal of the bundles after NDA verification by using different tools and following alternate paths. The cameras verify also the underwater movements of the empty and loaded baskets in the loading station area.

A third camera mounted directly above the basket replaces the NDA system performing simple item counting in the facilities where re-verification at the moment of basket loading is not required.

2.2. Step 2 – Transfer of the basket to the dry storage facility

Once a basket has been fully loaded and covered with its lid, the CoK over the basket during the further processing is maintained by using a battery operated small device consisting of a neutron detector assembly and a commercially available data logger (Madge Technology mod. Pulse101). This unit, denominated Mobile Unit Neutron Detector (MUND), is mounted in a waterproof enclosure fixed and sealed on the upper part of the transfer flask. The MUND has proven very effective in monitoring the movement of a loaded basket. As figure 1 shows, the MUND signal records the lifting of the basket into the drying station (1), the translation on the conveyor to the welding station (2), and finally the lifting of the welded basket into the transfer flask until removal when discharged into the concrete silo (3). Due to its peculiar pattern, the detected neutron signal cannot be tampered with, unless major modifications of the transfer flask are performed that would be spotted during regular DIV inspections.

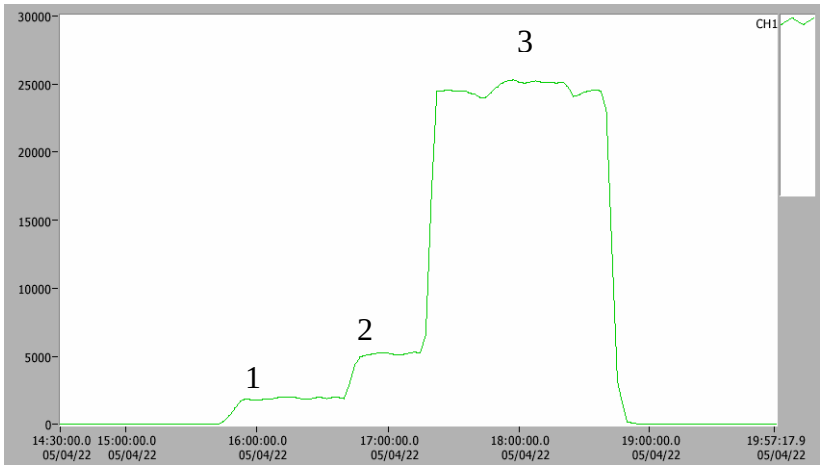

Figure 3 – Signal pattern recorded by the MUND during the transfer of a basket

The MUND can operate without service for more than 8 weeks. Servicing is quick and consists in swapping the unit with a spare one. Data are gathered from the removed unit by the inspector in the office by using a laptop computer connected through a serial interface adapter. After full battery recharge (6 hours) the unit is ready for reuse. Thanks to the small overall dimensions (30 x 20 x 10 cm) and weight (15 kg, including battery), installation is not intrusive and does not pose major problems to the operator. Possibility of remote data transmission directly from the unit mounted on the flask is under consideration for future implementation.

MUND reliability has been addressed by adopting internal redundancy of the data loggers and extensively testing in an environmental chamber with temperature excursion between -10 and 40 degrees C.

Backup in case of failure of a MUND unit is provided by installing 2 units to guarantee full redundancy or, alternatively, by video surveillance installed on the truck while the transfer flask is loaded and by the underwater cameras installed in the Spent Fuel Bay while the transfer flask is resting on the welding station.

2.3. Step 3 – Loading of the silo

The last step of the transfer process, the loading of a full basket into the concrete silo, is monitored by using a Silo Entry Gamma Monitor (SEGM) constituted by a pair of gross gamma silicon detectors inserted at different levels into the verification tubes available for each silo. Silos are loaded from the top and the sequence of the signals from the two detectors allows the verification of the direction of movement. The lower detector is mounted at such a level as to produce a steady signal when the last basket is loaded into the silo and the silo is full, hence behaving as an active seal. Several silos are equipped with SEGM detectors before the transfer campaign starts. The detectors can easily be relocated to empty silos by inspectors when the permanent sealing of a full silo is completed.

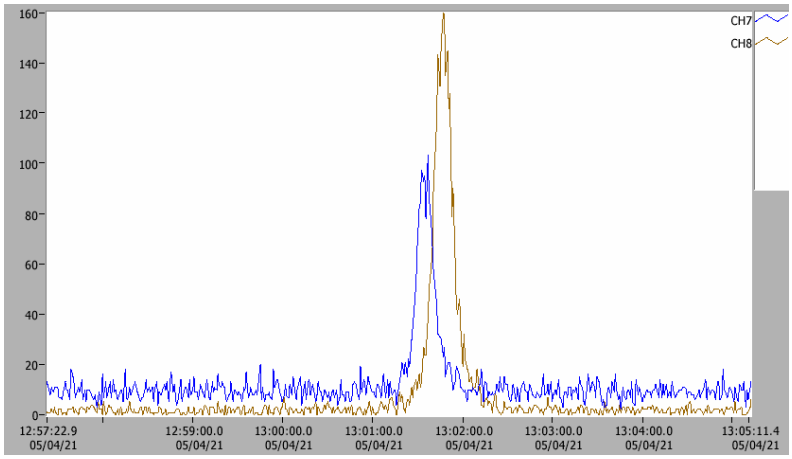

Figure 4 – Signal patter obtained while storing a basket in the concrete silo.

The electronics collecting the signal from all the detectors is located in a weatherproof cabinet located at the base of the silos. This cabinet is mounted on wheels to allow relocation as the campaign proceeds. A wireless communication link to a fixed entry point has been foreseen to allow implementation of remote data transmission.

Video surveillance provides backup in case of failure of the SEGM: depending on site-specific characteristics, two or more cameras cover the area of the dry storage facility interested in the campaign and monitor the loading operations being performed. Agreement between the images and the time-stamped radiation data from the MUND constitute a strong evidence that could replace the results of the SEGM in case of failure.

3. Field test results.

The system described in the previous section has been tested during 2004 transfer campaign in Cernavoda NPP - Romania. During this field test the system has operated in parallel with the standard inspector-attended approach. The results have been positive: the system was able to successfully monitor the transfer of more than 2500 bundles (~ 21 SQ of Pu) without any major equipment problem. Installing and servicing procedures and software have been tuned and finalized for optimal operation of the system. The system will be used in unattended mode during next transfer campaign.

In January 2005 the same system has been installed in one unit in Wolsong NPP – Republic of Korea for evaluation. After a positive completion of this evaluation period, the installation in the four units of the station is foreseen in the next months.

4. Conclusions

The implementation of a reliable unattended system to monitor the transfer of irradiated fuel has proven to be a valid substitution to inspector's constant presence during the transfer process, thus relieving inspectors of repetitive sealing and unsealing and NDA verification activities. The cost saving for the Agency is significant, with a foreseen reduction of 80% of the actual PDI spent for these activities. Besides, the operator could also benefit from the installation of an unattended system by saving the time spent in supporting the inspector's activities.

Although designed under a traditional safeguards regime, this approach can be implemented in countries going into an integrated safeguards regime. At the moment,
the integrated safeguards approach for the transfer to dry storage in Candu 600 has not yet been finalized. Most of the components described above will be likely reused under an integrated safeguards regime, complemented by an adequate mailbox system for operator's declarations, remote data transmission, and a scheme of random unannounced inspections.

Towards the Integrated Review of Safeguards Surveillance Data

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Abstract:

The increased use of surveillance systems, including unattended remote monitoring, results in a large amount of data to be reviewed by inspectors at headquarters. The review task is made difficult by the size and the variety of the data streams. There is thus interest in the development of tools for a 'Safeguards Review Station' (SRS) aiming at easing the interpretation of incoming and archived Safeguards data.
To date, several tools support the review only of individual sensor data (e.g. images, radiation measures, etc.). The CRISP review module of the RADAR system is an exception, as it provides a time-integrated platform to review nucleonic data streams collected at several points of interest. However, CRISP is limited to radiation sensors and the detection of one event by one sensor is independent of what the other sensors are 'suggesting'.
In this context, we are following a two-step approach for the development of the SRS.
Firstly, and when possible, we improve existing filters associated to individual sensors to detect Safeguards-relevant events with higher precision. Along this line, this paper presents such an improvement over an image review filter: a state-of-the-art scene change detection algorithm is augmented with two different search-by-content techniques applied to images to detect precisely typical classes of Safeguards-relevant events. We present test results on real Safeguards images.
Secondly, we propose a novel way to review all sensors' data in an integrated way based on a Hidden Markov Model framework. The core idea is that, while the detection of events starts from the individual sensors in parallel, this first layer of detection can be 'revised' by the SRS by considering the plausibility of the sequence of events detected by the collection of sensors. This plausibility check would be based on prior knowledge of typical sequences of events taking place in a given plant. This knowledge could be derived, for instance, by analysing archives of past review reports for a given plant.

Keywords: multi-sensor surveillance; integrated data review; Hidden Markov Models.

1. Safeguards reviews of surveillance streams

Goal of reviews. Safeguards reviews verify activity declarations prepared by operators of nuclear installations. To certify the completeness of declarations, nuclear inspectors review surveillance data gathered by sensors placed in these installations. In addition the entire period is also reviewed to confirm that no undeclared movements (which may indicate an attempted diversion of nuclear materials) take place.

Heterogeneous sensor data. Sensor data is from two categories:

- images, captured by time-lapse cameras;
- nucleonic data, measured by radiation sensors.

The practice at DG TREN I is that reviews of images and reviews of nucleonic data are performed *independently*. However, it is recognized that reviewing all surveillance data in an *integrated way* would facilitate the reconstruction of a plant's activity.

State-of-the-art on integrated data reviews. The development of the RADAR system [1] is a step in this direction: during data review, the inspector can see and correlate nucleonic data gathered by several sensors in a single view. A second example is a demonstrator that links images acquired by the FAST surveillance system with the NEUTRINO software for the detection of radiation events [2]. A third example is the KNOWLEDGE GENERATION software developed at Sandia National Laboratories [3]. The core idea is to use Finite State Machines (FMS) to model a plant's normal behaviour: sensors' data is then processed and matched to the FMS to confirm normality or to detect deviations. An application of this method to the Joyo remote monitoring system in Japan is presented in [4]. With a similar aim, [5] proposes a prototype intelligent monitoring system with a core expert system module that performs analysis of data and sequences of events to detect inconsistencies or abnormal operating patterns. The expert system knowledge base codes a number of heuristics developed from interviews with material control and accounting professionals. More recently, [6] describes a framework called OPTION (OPtimization Tool for the Instrumentation of Observation Networks) to verify, analyse, optimize and deploy networked sensors in view of detecting specific events. In OPTION, cost-effectiveness of the sensor configuration also plays a role.

Safeguards Review Station. The 'Safeguards Review Station' (SRS) presented in this paper shall be another contribution to the integrated presentation and analysis of heterogeneous sensor data.
We are following a two-step approach for the development of the SRS.
Firstly, and when possible, *we improve existing filters associated to individual sensors* to detect Safeguards-relevant events with higher precision. Along this line, this paper presents such an improvement over an image review filter: a state-of-the-art Scene Change Detection algorithm is augmented with two different search-by-content techniques applied to images to detect precisely typical classes of Safeguards-relevant events. We present test results on real Safeguards images.
Secondly, we propose a novel way to review all sensors' data in an integrated way based on a Hidden Markov Model framework. The core idea is that, while the detection of events starts from the individual sensors in parallel, this first layer of detection can be 'revised' by the SRS by considering the *plausibility of the sequence of events detected by the collection of sensors*. This plausibility check would be based on prior knowledge of typical sequences of events taking place in a given plant. This knowledge could be derived, for instance, by analysing archives of past review reports for a given plant.

2. Tools for image reviews

Two software prototypes are being developed to assist nuclear inspectors in the review of surveillance images. *These tools are meant to extract all Safeguards-relevant images out of surveillance streams acquired by time-lapse cameras: the goal is to speed-up the review process.*

Scene Change Detection. To date, image reviews at DG TREN I are facilitated by the use of GARS [7]. GARS performs a reduction of the dataset to be reviewed by means of a Scene Change Detection (SCD) algorithm. Typically, SCD measures the change in grey values between temporally consecutive stored images in user-defined Areas of Interest (AOIs). When the change exceeds programmed thresholds, the corresponding images are selected for inspector's review.

Search-by-content technologies. In our review prototypes, SCD is followed by the use of informed *search-by-content technologies.* These operate on an abstract representation of the image called the image *feature description* [8]. For instance, a black and white image can be described by a *grey histogram* of 256 components, where each bin in the histogram is a count of the number of pixels with a specific shade of grey. Alternative representations may include *shape factors* of segmented image regions (like the circularity or the compactness of the regions), or one can combine in a single representation grey and shape features. The range of possible image representations is vast. While choosing one, we keep in mind that: (i) a compact feature description limits the computation necessary to manage and subsequently search for relevant images in large surveillance sets; (ii) the feature description must preserve the *information content* that will allow an automatic search technique to decide if an image is relevant or irrelevant. To ascertain that this is the case, experimental confirmation is needed.

Figure 1: Time-line view of the classified image stream: yellow squares stand for images classified 'relevant'.

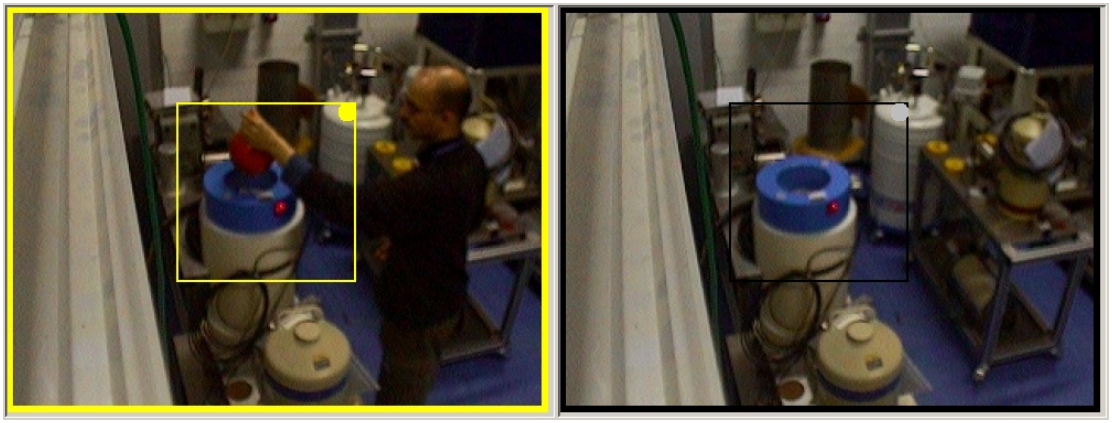

Figure 2: An image classified 'relevant' (left) and one classified 'irrelevant' (right) are shown on the viewer for inspection. The colored frames (yellow or black) around the image and the AOI indicate the image classification.

Image retrieval and image classification. Two search-by-content techniques were investigated: *image retrieval* (IR) and *image classification* (IC).
With IR [9], the inspector selects an *example image* representing the event s/he wants to find in a given image stream: IR extracts from the stream filtered by SCD all images that are similar to the chosen example. The similarity of one image to the example is given by the inverse of the Euclidean distance between two feature vectors, one describing the image, the other describing the example: the shorter the distance, the higher the similarity.
With IC [10], the inspector selects *a set of training images* including both examples of relevant and irrelevant images. This training set is used to build a classifier that learns to attach a relevant or irrelevant label to each image in the stream filtered by SCD on the basis of its features' values: images labelled 'relevant' will be subsequently reviewed with priority.

Relevance feedback. In addition, we have augmented the basic IR and IC approaches by a *relevance feedback* mechanism [10]: the inspector is given the possibility to provide feedback to the system about its retrieval/classification performance to refine the search results. This is part of the training or set-up of the tools and, in particular, it supports the automatic selection of a suitable feature representation for the images among a repertoire of possible ones.

Time-line view of the image set. Finally, the review is made easier at interface level by a *time-line view* [4] of the classified image stream: images are represented on a line as time-ordered squares that are highlighted in yellow when they correspond to images classified 'relevant' (Figure 1). The time-line preserves the *temporal context* of images and allows a fast search of relevant scenes in the complete stream: for example, moving the mouse on the line loads the corresponding images on a viewer for inspection (Figure 2).

Safeguards image sets. In the following, we focus on experimental results obtained by running these image review prototypes on black & white images acquired by a GEMINI surveillance system [11]. The tests, performed at DG TREN I headquarters, aimed at the detection of two types of events in a Material Balance Area (MBA):

- E1 - flask visible over hatch (entry or exit);
- E2 - flask visible over pond (entry or exit).

Image set	Images	Events E1	Events E2	All events
IS1	16020	30	32	62
IS2	15446	11	12	23
IS3	20179	17	17	34
IS4	15661	1	1	2
IS5	16022	-	-	-
IS6	20166	-	-	-
IS7	15668	-	-	-
IS8	20141	-	-	-
IS9	15847	-	-	-

Table I – Image sets characterization

We ran our systems on 9 image sets taken in two different MBAs. The first two image sets, IS1 and IS2, where taken in MBA A; 7 image sets, IS3 to IS9, were acquired in MBA B. Each image set spans a period of 3 months approximately. This means, for example, that there is time interval of 18 months between IS3 and IS9.
Table I shows the number of images and events contained in each image set: this ground truth information has been derived from the archived reports prepared by inspectors at the time they reviewed of these image sets.

Tests. We reviewed the 9 image sets using three combinations of techniques:

a. SCD;
b. SCD followed by IR;
c. SCD followed by IC.

Figure 3 shows in a schematic way how we combine the different techniques. Given the stream of images, the first step is to apply the SCD filter. This takes two consecutive frames and checks whether the change in the last one exceeds the detection threshold. If not, then the image is classified as irrelevant. Note that in this case we do not need to run the IR or the IC steps, because we trust the decision taken by the SCD filter. On the other hand, if the change exceeds the detection threshold, we can accept the SCD decision and classify the image as relevant (a.), or submit the image to the IR (b.) or IC (c.) filters for a second opinion.

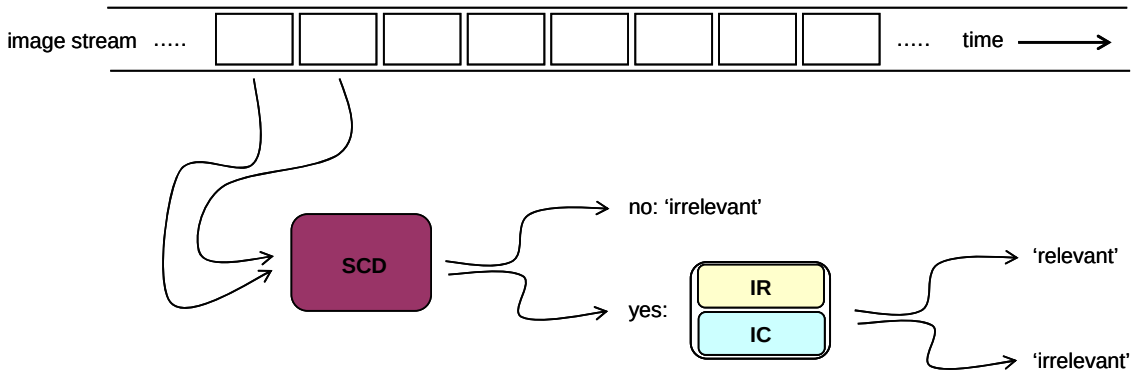

Figure 3: Schema showing how the SCD filter is combined with IR or IC to classify the images in the stream.

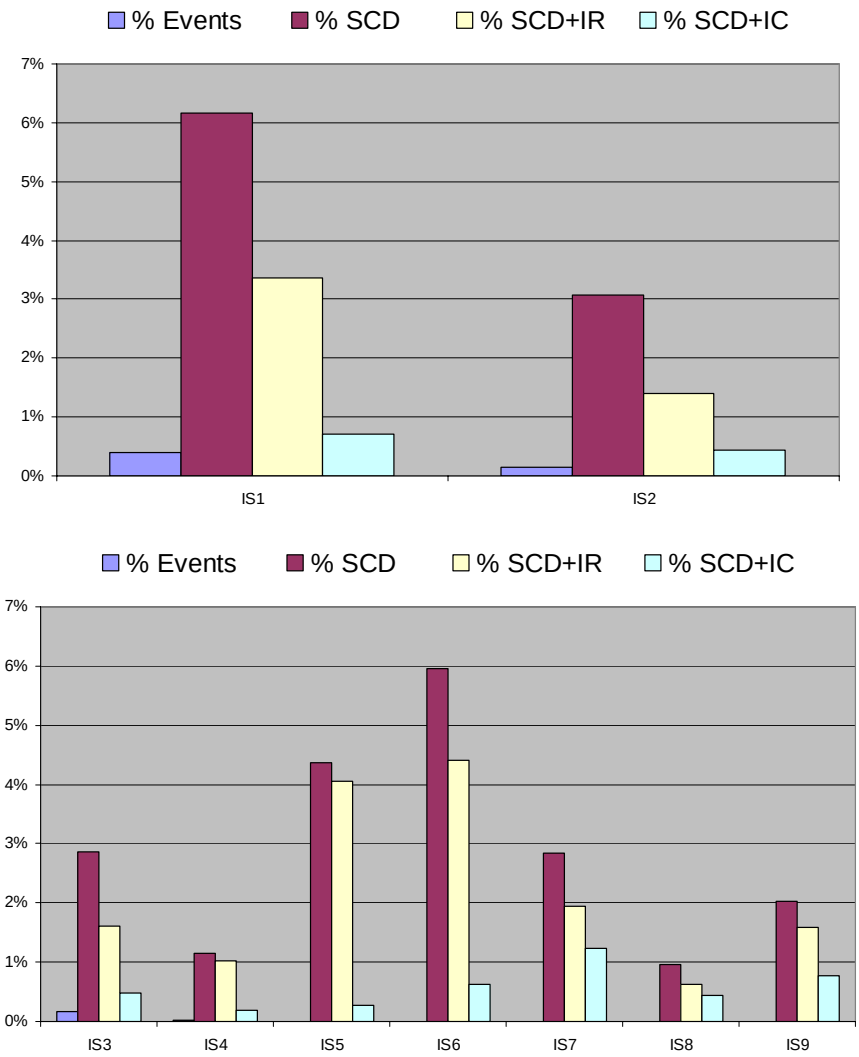

Figure 4: For MBA A (upper plot) and MBA B (lower plot) and each image set, IS1 to IS9, a group of four bars measure (as a percentage of each image set): the images that relate to safeguards-relevant events (blue bar); the images proposed for review by SCD (purple bar); the images proposed for review by SCD followed by IR (yellow bar); the images proposed for review by SCD followed by IC (light blue bar).

Performance metrics. The absolute performance requirement for an image review tool is that it *detects all Safeguards-relevant events*. In addition, it is interesting to *quantify the tools' ability to present to the inspector's attention parts of the image stream containing only Safeguards-relevant events*.
Therefore, the evaluation of the performance of an image review tool is based on checking:

- if relevant images have been missed;
- how many irrelevant images are proposed for review.

Results. Firstly, we can report that, with all three review techniques, no relevant event of type E1 and E2 has been missed. Secondly, regarding the number of images proposed for review, the results are summarized in Figure 4 for MBA A (upper plot) and for MBA B (lower plot). In these plots, for each image set, the four bars measure (as a percentage of the entire image set):

- the images that relate to safeguards-relevant events (blue bar);
- the images proposed for review by SCD (purple bar);
- the images proposed for review by SCD followed by IR (yellow bar);
- the images proposed for review by SCD followed by IC (light blue bar).

In other words, the number of irrelevant images detected by each technique is obtained by subtracting the blue bar from the each of the others.

Discussion. All three review tools tested in these experiments were able to detect all the events 'flask over hatch' and 'flask over pond' contained in the image sets used as benchmark, but with a difference in performance. Looking at the number of irrelevant images proposed for review, these experiments show that coupling IR/IC search-by-content technologies with SCD pays off. The improved performance is due to the amount of information these techniques rely on to select the images from the SCD filtered stream. While SCD is triggered by any change (above a programmed threshold) between consecutive images, IR/IC are targeted search-by-content techniques: they make reference to example images labelled as relevant by inspectors. In other words, IR and IC learn from visual regularities in nuclear processes, while SCD is totally neutral in this sense.
For all three review tools, one important fact to stress about their set-up or training is that, though for each considered MBA the systems have been trained with a small set of images, their performance extends well to other datasets acquired in a different time period in the same installation
For the IR tool, the training phase was oriented towards the guarantee that all the relevant images were indeed proposed for review. This concern increased the number of false positives and influences the correct classification rates. However, one could say that, for IR, the possibility of setting a more or less conservative similarity threshold is a positive property of the algorithm in that it allows, if so is wished, for a direct user control. This property is also shared by the SCD algorithm. For the IC tool, which in these experiments achieved the best performance, the control on the detection is indirectly in the hand of the user through the selection of the training examples, because the classifier is built automatically.

3. Integrated reviews

In the previous section we have addressed the review of a stream of surveillance data generated by a single sensor (a camera). The aim was to detect different classes of events ('flask over hatch' and 'flask over pond') from a unique source of images. To recognize these classes, we have used 2 AOIs defined on the image plane and a search-by-content technique tuned on each of these. These adaptive filters act as 'virtual sensors' for the target classes of events. Thus, these experiments can be framed as a multi-sensory classification context.
In the current set-up, each virtual sensor decides completely independently from the others. This is a limitation of the design, because the classes of events to be detected tend to appear in structured sequences due to regularities in the nuclear process. Thus, we argue that the detection of one event by one virtual sensor, especially if confirmed by the inspector at review time, should have an influence on what the other virtual sensors are detecting.
This also applies to truly multi-sensor contexts (e.g. several cameras) or in networks of heterogeneous sensors where one needs to mix and integrate information coming from radiation sensors with images.

Sequences of safeguards-relevant events. To illustrate the nature of the sequences of events, Figure 5 shows a report produced by inspectors as the result of the review surveillance images. For a given MBA, the report lists chronologically all events observed by the inspectors in the surveillance stream: the events describe the 'actual' nuclear material flow within the MBA and this will be compared with the declaration provided by the plant operator.
Images of interest annotated by the inspectors record the events 'flask over hatch' (H), 'flask in decontamination' (D) and 'flask over pond' (P): a single flask enters over the hatch, it stays in decontamination area, then it goes into the pond; from the pond, it goes back to decontamination and finally it exits from the hatch area. Thus, for this MBA, the expected sequence of events is H-D-P-P-D-H.

Non-deterministic sequences. In reality, the pattern of events is non-deterministic. As an example, the sequence of events in the review report of Figure 5 reads as:

… H-D-H-D-P-P-D-H-H-D-H-H-D …

The variability in the order of the events may be due to different reasons. Firstly, it can be the case that a flask is seen entering over the hatch, then goes immediately back without undergoing the expected sequence H-D-P-P-D-H: one may wonder why the process was stopped, but nevertheless this sequence of events may be valid from the Safeguards point of view. Secondly, certain MBAs allow for more that one flask to be in decontamination area at the same time.

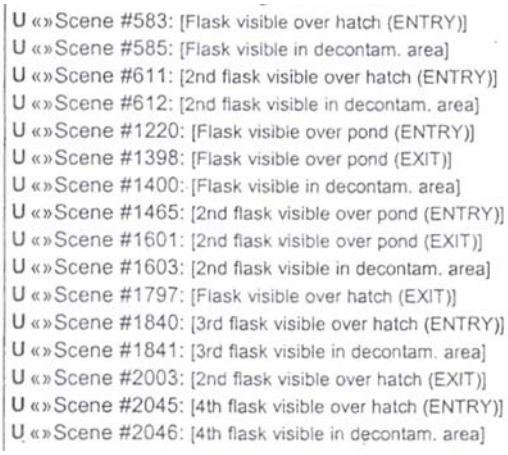

Figure 5: Review report produced by inspectors.

As a consequence, several flasks can be seen entering over the hatch and are just stocked in decontamination before one of them is moved to the pond. Also, these several flasks staying in decontamination are undistinguishable from the surveillance images alone: the inspector can only verify how many flasks moved in and out of an MBA.
In conclusion, although the nuclear process can be thought as deterministic, the actual process looks as a stochastic one. Therefore, we should look for modelling techniques that may account for this natural and valid variability.

Hidden Markov Models for the detection of sequences of events. The problem of recognizing a sequence of Safeguards-relevant events in an MBA is equivalent to that of solving a *decoding problem* in a *Hidden Markov Model* (HMM) [12].

Applications of HMMs. As a background, HMMs have been applied successfully to several pattern recognition problems, such as the discrimination of spoken language [13] or handwriting recognition [14]. What is common to these applications is that there is an observable ('measurable') sequence of signals, e.g. sounds for speech and strokes for handwriting. These signals relate probabilistically to some 'not-directly measurable' entity: in speech, sounds relate to phonemes, in handwriting, strokes may relate to letters. In turn, the non-observable, 'hidden' entities are connected by some statistical relationship. For example, in a given language, certain sequences of letters are more likely than others. This statistical structure can be modelled and used to assist the automatic recognition of spoken words. HMMs provide a suitable framework for this enterprise.

Schema of an HMM. Figure 6 shows schematically an HMM. The circles stand for the hidden states and the arrows between them indicate possible transitions between states. These links are labelled by probabilities (not shown) which form the so called transition matrix. On the lower part of Figure 6, we represent the observable states as squares that contain sensor readings. The observable states are linked to the hidden states. The probabilities associated to these links (not shown) form the confusion matrix.

HMM and MBAs. Generally speaking, a HMM is used to describe a system that changes state in a probabilistic way. For example, an MBA can be seen as a system where the Safeguards-relevant events are its possible states: the sequence of Safeguards-relevant events that takes place in the MBA is a probabilistic sequence.

MBAs and the Markov assumption. In an HMM, the model is markovian: the state of the system depends only on previous states, not on external factors. The Markov assumption may not be fulfilled in the case of MBAs. This is because we may decide not to model what causes the flow of a flask in the MBA (e.g. the movements of cranes). Nevertheless, the Markov assumption simplifies the analysis of the system and the identification of the system states (i.e. the recognition of Safeguards-relevant events) may still be accurate enough. Actually, we can report that most successful applications of

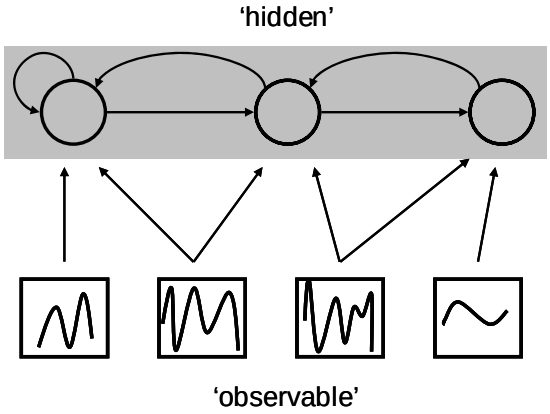

Figure 6: Hidden Markov Model.

HMM do violate the Markov assumption, but modelling results are viable. For example, for handwriting recognition, to be precise we could model the movement of the hand as a factor that contributes to the stroke shape; on the other hand, if we just concentrate on the strokes produced and the sequences of detected letters, we have a simpler system that can work pretty well.

MBAs and the stationarity assumption. In an HMM it is also assumed the probabilities of transition between states remain stable over time. In the case of an MBA, this assumption should be met, unless the underlying nuclear process changes completely, a case which should be verifiable.

MBAs and hidden states. The Markov model is said to be 'hidden' because its state is not directly accessible, but it is related probabilistically to some observables. Likewise, nuclear inspectors do not see Safeguards-relevant events as they take place in an MBA, but rely on sensor readings (e.g. images and radiation signals) which relate probabilistically to the events of interest. For example, in a multi-sensor context, an image of a flask over the pond together with a radiation peak is a strong hint for a flask-over-pond event. A radiation peak over the pond is a weaker hint, but is still an indication of a flask-over-pond event.

Elements of an HMM. As a summary, an HMM is described by (a) a set of hidden states, (b) the transition probabilities between the states (transition matrix), (c) some observable states and (d) the probabilities that relate the observable states to the hidden states (confusion matrix). For an MBA, (a) and (c) are given, respectively, by the nuclear process and surveillance sensor configuration; on the other hand, (b) and (d) must be learned (estimated) on the basis of historical data. For example, the transition matrix can be learned by scanning archives of review reports produced by inspectors for a given MBA. Likewise, the confusion matrix could be learned by time-correlating events annotated in the review report with various sensor readings.

The decoding problem. Having instantiated an HMM, the decoding problem is to determine the sequence of hidden states that most probably has generated an observed sequence. The decoding problem is solved by the classical Viterbi algorithm [15]. Coming back to MBAs, we can address the Safeguards review problem by applying the Viterbi algorithm to an HMM that describes the MBA. Thus, given a sequence of sensor readings, we can estimate the sequence of Safeguards-relevant events that most probably produced those sensor readings.

4. Conclusions

We have presented the current baseline of the project Safeguards Review Station, whose general aim is to develop new tools to assist inspectors in the review of surveillance data gathered from installations.
Its first goal is to attach accurate filters to individual sensors, that is, to perform a first reduction step on the massive amount of data produced by each sensor to highlight the interesting events. The first part of this paper has presented some experimental results obtained on image reviews. These show that to search for specific classes of safeguards-relevant events, one can augment a state-of-the-art Scene Change Detection filter with more informed search-by-content technologies (Image Retrieval and

Image Classification): the effect is that the detection of the relevant events is preserved but with far less false positives.
The further goal of the project is to design and evaluate an architecture for a truly integrated review, where the multi-sensor perspective is taken into account. In the second part of this paper we have sketched the possibility of modelling Material Balance Areas by means of Hidden Markov Models. The opportunity of this choice is indicated by the fact that, even when the procedure underlying an MBA are deterministic, their execution opens the door to a certain variability that is compatible with what can be considered a normal operation pattern. By far the most difficult part in applying the HMM modelling technique is in the estimation of the model parameters, that is, the transition and the confusion matrices. These matrices must be learned of the basis of archives of historical data: review reports and the corresponding sensors' readings. Once the HMM is estimated, the problem of reconstructing an MBA activity is quite straightforward using the Viterbi algorithm.

5. References

[1] Schwalbach, P., Chare, P., Girard, T., Holzleitner, L., Jung, S., Kloeckner, W., Roesgen, E., Smejkal, A., Swinhoe, M.; *"RADAR": The Standard Euratom Unattended Data Acquisition System*; Proc. of IAEA Symposium on International Safeguards; 2001.
[2] Heppleston, M.; Personal communication; 2003.
[3] DeLand, S.M., Brabson, J.M., Smith, J.D., Jaramillo, T.I., Spaven, S. M.; *Analysis of Unattended Monitoring System Data Using Knowledge Generation* Software; Proceedings of the Institute of Nuclear Materials Management; 2000.
[4] Damico, J.P., Hashimoto, Y.; *Automated Analysis for the Joyo Remote Monitoring System* ; Proc. of the Institute of Nuclear Materials Management; 2002.
[5] Laurin-Kovitz, K.F., Ewing, T.F.; *Intelligent Monitoring and Analysis for Safeguards Applications*; Proceedings of the Institute of Nuclear Materials Management; 2001.
[6] Garcia, H.E., Yoo, T.; *OPTION: A Software Package to Design and Implement Optimized Safeguards Sensor Configurations*; Proc. of the Institute of Nuclear Materials Management; 2004.
[7] Canberra Aquila; http://www.aquilagroup.com/pdf/canberra/GARS.pdf.
[8] Del Bimbo, A.;*Visual Information Retrieval*; Morgan Kaufmann Publishers; 1999.
[9] Stringa E., Meylemans P., Gonçalves J.G.M.; *A Safeguards Review Station Prototype based on Image Retrieval Tools*; Proc. 25th ESARDA Symposium on Safeguards and Nuclear Material Management; 2003.
[10] Versino C., Stringa E., Gonçalves J.G.M.; *A Filtering Tool to Review Surveillance Image Streams*; Proc. 25th ESARDA Symposium on Safeguards and Nuclear Material Management; 2003.
[11] Canberra Aquila; http://www.aquilagroup.com/pdf/canberra/Gemini_Serveillance_Sys.pdf.
[12] Rabiner, L.R., Juang B.; *An Introduction to Hidden Markov Models*; IEEE Acoustics, Speech & Signal Processing Magazine; 3; 4-16; 1986.
[13] Rabiner, L. R.; *A Tutorial on Hidden Markov Models and Selected Applications in Speech Recognition*; Proceeding of the IEEE; 77 (2); 1989.
[14] Hu, J., Brown, M., Turin, W.; *HMM Based On-Line Handwriting Recognition*;. IEEE Transactions on Pattern Analysis and Machine Intelligence; 1996.
[15] Forney, G.D.; *The Viterbi Algorithm;* Proceedings of the IEEE; Vol. 61; No. 3; pp. 268 – 278; 1973.

Poster session

On-Line Verification of MOX Fuel Magazines Using Monte Carlo Simulation

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Abstract

The rationale behind using the Monte Carlo technique for the verification of MOX fuel elements has been thoroughly investigated and shown to be sound, reliable and perhaps the best way forward in view of the limiting conditions on site. A neutron coincidence counter, SMP-D4 installed at the Sellafield Mox Plant (SMP) for the verification of the flow of MOX fuel pins during production, has been fully modelled using the Monte Carlo codes, MCNP-PTA, which simulate both the neutron transport and the coincidence electronics. 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 extremely well with our 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.

Keywords: Neutron counting, calibration, NDA, Monte Carlo simulation, MOX fuel

1 Introduction

Various neutron coincidence counters have been permanently installed at the BNFL Sellafield MOX Plant (SMP) in order to verify the flow of safeguarded nuclear material during production. Of interest to this work is a neutron coincidence collar named SMP-D4 [1-3] used at the later stages of the process whereby the fuel pins are loaded into magazines, a matrix representing the layout of the fuel assemblies.

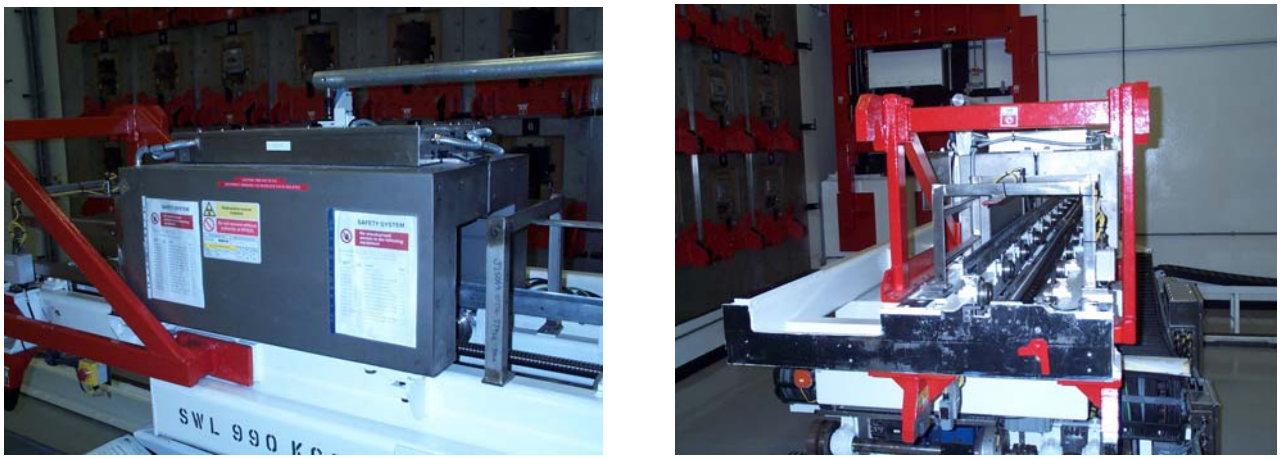


Figure 1: Photos of the SMP-D4 NCC counter with the magazine handler with BNFL kind permission.

Due to the unavailability of suitable calibration standards, a procedure for the numerical calibration of the detector has been applied. The MCNP-PTA codes, described in references [4-6] was developed at JRC, as an extension to MCNP4-C2 [7] and allows the simulation of both neutron transport and coincidence electronics in order to calculate the response of neutron coincidence counters.

From the operational point of view, the main difficulties derive from the fact that:

- The magazines to be verified may contain only intermediate rod configurations corresponding to a partial loading rather than a full one of fuel elements.
- The efficiency profile inside the detector cavity show strong variation.
- Access to the area is very restricted.

The solution adopted, based on the Monte Carlo radiation transport technique, has proven to be reliable when applied to similar problems [8-10], various detector systems and radiation fields. In essence, it allows the inspector, on site, to run MCNP-PTA code for a given pre-declared loading pattern from which the Reals and Totals rates are obtained.

In this paper, we report on recent work carried out to validate the Monte Carlo model of the SMP-D4 detector using a set of measurements carried out on site with a number of well known MOX fuel rods. The rationale and motivation behind our approach is discussed. We also describe a user interface which allows the inspector to prepare for and easily run the Monte Carlo calculation for a pre-declared fuel element loading pattern. The user interface allows a simple and fast definition of the loading pattern of the magazine. The MCNP-PTA input file is automatically generated coupling a fixed detector model with a variable fuel model. The Monte Carlo calculation can be run on-line and the inspector can verify whether the measured and the calculated count rates are consistent.

2 The Monte Carlo Simulation Model

The SMP-D4 is an open ended customised version of the neutron coincidence collar (NCC), made of three separate high-density polyethylene moderators encased in cadmium and stainless steel and within which a total of thirty ^{3}He proportional counters are embedded. The reaction involved with incoming neutrons is thus $^{3}He(n,p)^{3}H$. Details regarding counter design, various performance tests, system parameters determination have been previously described by Croft [1-2] and by H. Tagziria [3] and thus will not be repeated here. A detailed modelling of SMP-D4 neutron coincidence counter was carried out using MCNP-4C2 [9] and the geometry was formulated from drawings of the constructional details of the counter that have been obtained from Harwell Instruments (UK).

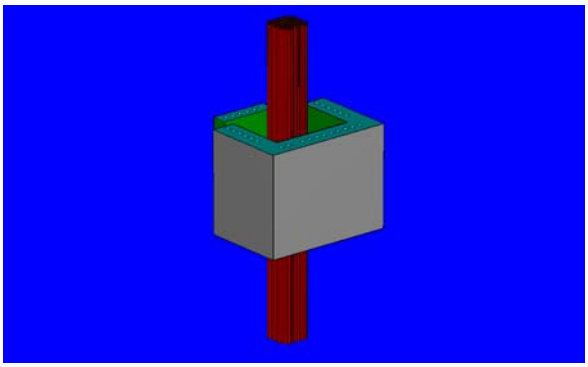


Figure 2: Model of the D4 counter with fuel element rods

However, a simulation of the neutron coincidence counting and the pulse train analysis that safeguards the pulse timing information (pulse trains) is further required if the Monte Carlo based method is to be used for verification purposes. This is achieved using MCNP-PTA Monte Carlo code, developed at the Joint Research Centre in ISPRA (Italy) as an extension to MCNP as previously described [4-6]. In essence, this code runs in

two phases. First the neutron transport simulation is performed in an analogue mode using MCNP and a pulse information file (PIF) is created where the pulse trains are banked. In the second phase, the analysis of the pulse trains is performed using the so called PTA code [4-6] for, as in the present case a shift register analyser, yielding the Totals and Reals rates being sought. The code can also analyse Time Correlation analysers and Multiplicity counters.

3 Validation of the model

Using the system parameters previously determined by Croft [1,2] and Tagziria [3], given in Table 1,

Gate	Pre-Delay	Die-away	Amplifier Dead time	OR-Chain DeadTime
64μs	4.5μs	69.7μs	1.21μs	50ns

the efficiency of SMP-D4 coincidence counter for a ^{252}Cf source was calculated by MCNP-PTA and found to be:

ε(simulated) = (11.07 ±0.03) % cm^{-2} which agrees well within 0.5 % with the measured value of

ε (measured)=(11.14 ± 0.04)% cm^{-2}.

In order to further validate our Monte Carlo simulation model as well as the approach followed, a set of 3 magazines containing about 100 known MOX fuel rods were measured on site at Sellafield. Figure 3 shows an example of Reals rates measured for a given fuel loaded magazine. The calculations which correspond to these measurements were carried out using MCNP-PTA, the results of which, as shown in Table 2, agree very well with the measured rates.

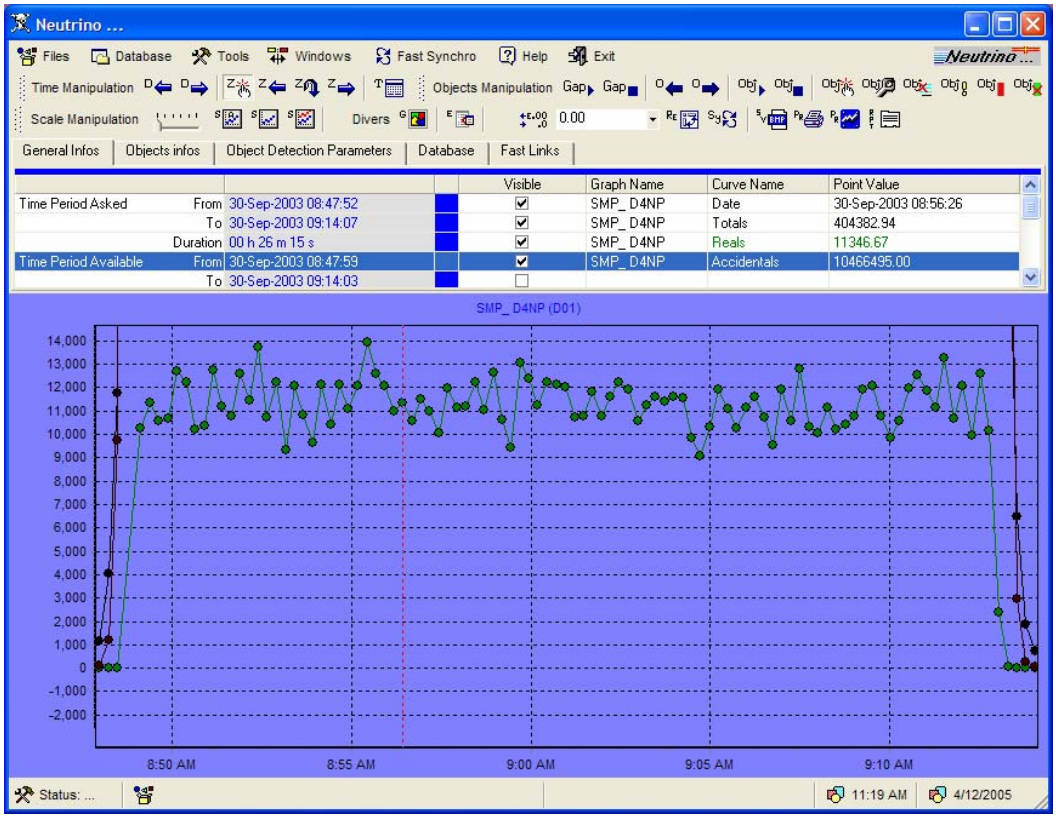


Figure 3: An example of measurements used for validation showing the Reals rates.

Measurement	A	B	C
Measured Reals (/s)	11328±996	11455±940	11197± 715
Calculated Reals (/s)	11396±593	11565±505	11159 ±422
Discrepancy (%)	0.6	0.1	-0.3

Table 2: Reals rates of MOX fuel pin magazines measured at Sellafield using SMP-D4 counter compared to MCNP-PTA calculations.

4 Rationale of the method and case study

It was said in the introduction that from the operational point of view, in addition to the fact that access to the counter is severely restricted, the main difficulty derives from the fact that the magazines to be verified may contain only intermediate rod configurations corresponding to partial loading rather than full fuel elements. Furthermore, constrained by mechanical design, which results in a non-symmetric detector geometry, the efficiency profile inside the detector cavity is strongly variable. Consequently, the response function does not only depend on the plutonium mass within the magazine but also on the geometry of the rod-loading pattern. It is hence neither possible to establish a classical calibration curve correlating mass to count rate, nor feasible to pre-calculate every one of the large number of possible configurations.

This is clearly illustrated in Figure 4 which shows result of the calculations carried out for a 14x14 MOX fuel pin magazine partially and progressively loaded from top and bottom. The Reals rates which are strong signatures of the mass of plutonium are different for the same mass depending whether the rods are in the lower part of the magazine or at its higher end.

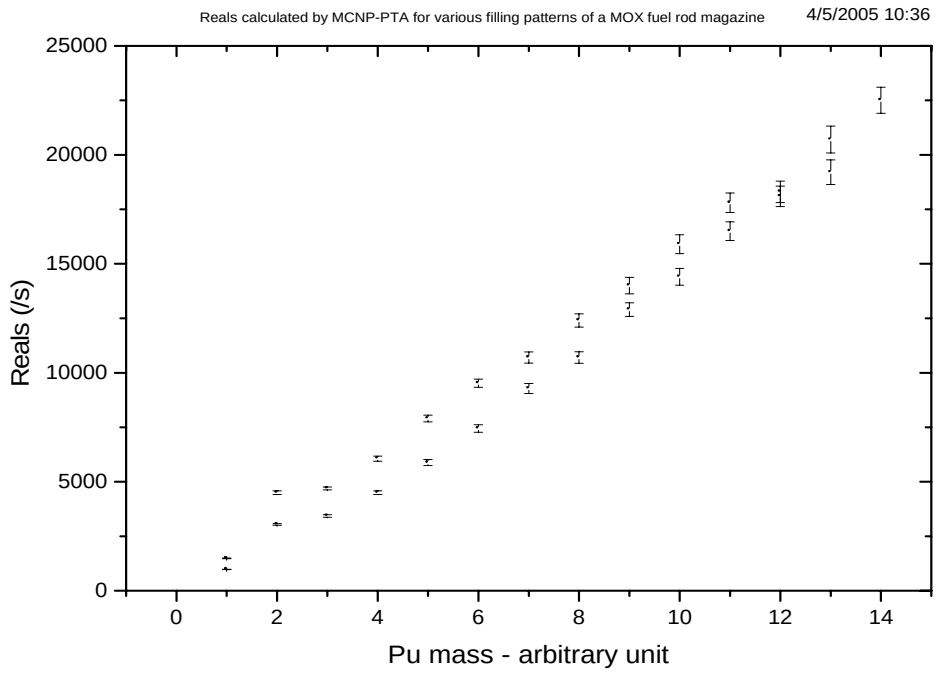


Figure 4: Reals rates calculated by MCNP-PTA for various patterns and masses a MOX fuel magazine.

5 User Interface

A user interface has been developed using the TCL/TK language [11] which allows a user to create the MCNP-PTA input file for the declared loading pattern and enrichments of the MOX fuel elements. Initially a base input file of the model containing details of the detector geometry and its materials and other necessary information is pre-loaded and the inspector may need only to change the loading positions (either individually with a simple mouse click or in group of rows and columns), the fuel type and isotopic composition (see figures 5-7), pin dimensions etc. At entry, the input data file management page (Figure 5) allows to read in the base input filename from a browser facility and to select the matrix structure of the magazine (e.g. 14x14 or 17x17). Figure 6 is the main page from which the remaining ones (figures 7) are activated. All changes are automatically written to a file prior to running the simulation code.

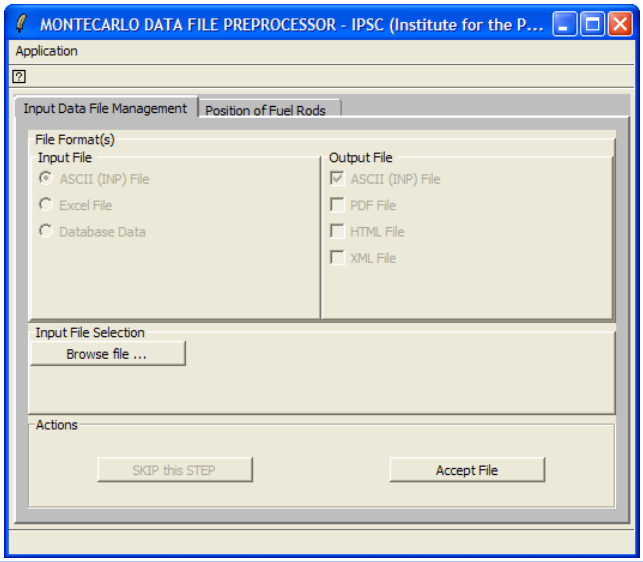


Figure 5: Entry Page

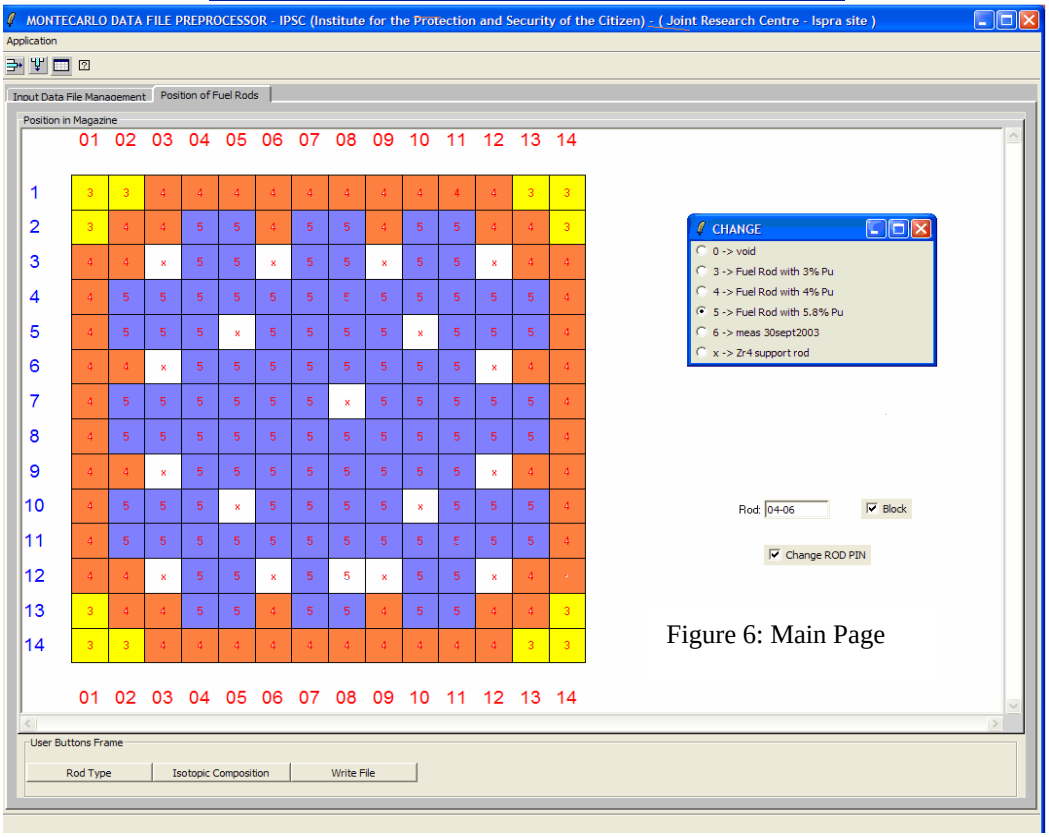
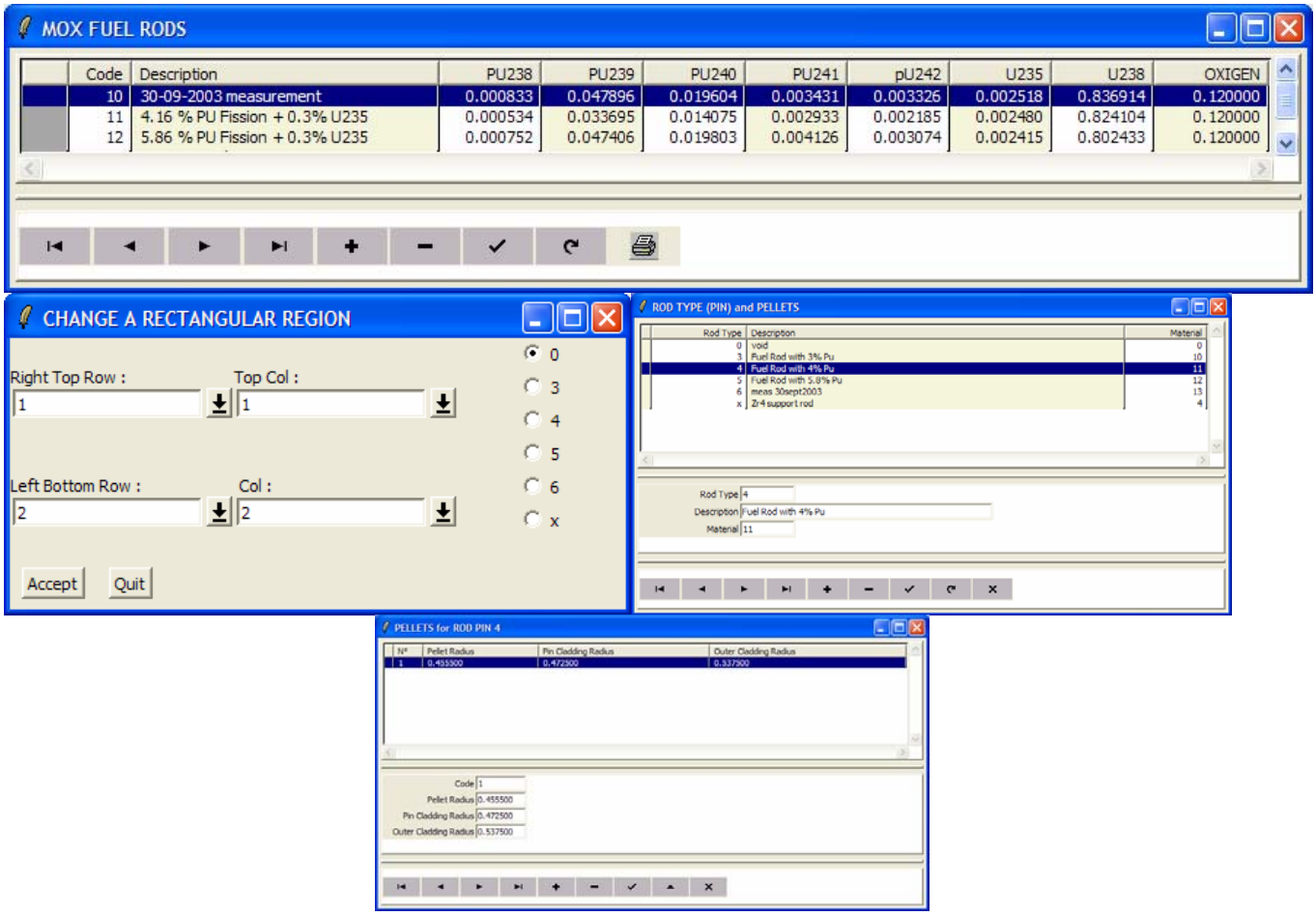


Figure 6: Main Page



Figures 7: Examples of pages activated from the main page of the user interface.

6 Summary and conclusion

The rationale behind using the Monte Carlo technique for the verification of MOX fuel elements has been thoroughly investigated and shown to be sound, reliable and perhaps the best way forward in view of the severely limiting conditions on site. The Monte Carlo simulation model of the BNFL SMP-D4 neutron coincidence counter has been successfully modelled and validated with MCNP-PTA Monte Carlo code including the pulse train analysis. The model includes the most important details of the counter as well as a full description of the MOX fuel magazines in any loading configuration and enrichment. Very good agreement was obtained between measurements and calculations both for MOX fuel elements in a magazine and for a radionuclide ^{252}Cf source. A user interface has been developed to simplify the preparation and running of the codes for the declared MOX loading pattern and enrichment. The model will be installed at the BNFL reprocessing plant (in Sellafield, UK) to be used as a verification tool during MOX production by the safeguards inspectors of the European Commission.

7 References

[1] S. Croft, P.M.J. Chard, I.G. Hutchinson, P.J. Price and P Schwalbach, ` Design and performance of neutron detector assemblies for a MOX fuel fabrication plant', Proceedings of the 19th Annual Symposium on Safeguards and Nuclear Material Management, Montpellier, France, May 1997, page 245.

[2] S. Croft, `User Guide for SMP Neutron Monitor D4', Harwell Instruments, AEAT 96/1011/D4/UG, March 1997.

[3] H Tagziria, P. Peerani, M Looman, and P Schwalbach, Monte Carlo Modelling of a Neutron Coincidence counter for the Verification of Mox Fuel Elements. Proceedings of the 25th Annual Symposium on Safeguards and Nuclear Material Management, Stockholm, Sweden, 12-15 May 2003

[4] M.R. Looman, N. Farese, G. Gonano, R. Jaime, B. Pedersen, and P. Schillebeeckx, "Monte Carlo Prediction of the Response of Neutron Counting Instruments", Proceedings of the 21th Annual Symposium on Safeguards and Nuclear Material Management, Seville, Spain, 4 – 6 May 1999, ESARDA Proceedings 29 (1999) pp. 375-381.

[5] M.R. Looman, P. Peerani, and P. Schillebeeckx, "An Overview of NDA Instruments Modelled with the MCNP-PTA code at the JRC Ispra", Proceedings of the 23rd Annual Symposium on Safeguards and Nuclear Material Management, Bruges,

[6] M.R. Looman, Proceedings of the 23rd Annual Symposium on Safeguards and Nuclear Material Management, Belgium, 8 – 10 May 2001.

[7] J Briesmeister (ed.), 'MCNP – A General Monte Carlo Code for Neutron and Photon Transport', *Los Alamos Report*, LA13709-M, Version 4C, Los Alamos National Laboratory, Los Alamos, 2000.

[8] P. Peerani, and M Looman, `Computational modelling of NDA instruments for nuclear safeguards', Proceedings of the 24th Annual Symposium on Safeguards and Nuclear Material Management, Luxembourg, 2002

[9] M Looman, P. Peerani and P Schwalbach, Calibration with Monte Carlo calculations of a neutron collar for the verification of FRM-II fuel elements', Proceedings of the 25th Annual Symposium on Safeguards and Nuclear Material Management, Stockholm, Sweden, 12-15 May 2003

[10] P. Peerani, P. Schwalbach, G. Bosler, I. Cherradi: "Computational Calibration of Neutron Collars for the Verification of HEU Fuel Elements", INMM 45th Annual Meeting, Orlando, Florida, (18-22 July 2004).

[11] John K. Ousterhout. *Tcl and the Tk Toolkit*. Reading, MA: Addison-Wesley, 1994 and http://www.lib.uchicago.edu/keith/tcl-course/tcl-course.htm

Source Term Analysis for Advanced Spent Fuel Conditioning Process (ACP) in Support of Safeguards Approaches

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Abstract:

The Advanced Spent Fuel Conditioning Process (ACP) is a pyrochemical processing technique to convert oxide-type spent nuclear fuel into a metallic form. The Korea Atomic Energy Research Institute (KAERI) has been developing this technology for the purpose of spent fuel management, and is planning to perform a lab-scale demonstration in 2006. With this technology, a significant reduction of the volume and heat load of spent fuel is expected, which could decrease the burden of final disposal in terms of disposal size, safety, and cost. In this study, source terms of the ACP materials were analyzed for the development of an efficient safeguards system that could meet international requirements. The goals were to investigate the energy spectra and event rate of gamma, (α,n), and fission neutron emission. Monte Carlo simulations were performed to investigate the variable materials in the process from a non-destructive (NDA) point-of-view. The results of this study will be useful for the development of optimized nuclear safeguards systems that can meet IAEA safeguards criteria.

Keywords: Pyroprocessing; ACP; Source Term; ORIGEN-S; MCNPX

1. Introduction

Since the fist commercial operation of nuclear reactors in the 1970's, approximately 6,000 metric tons of spent nuclear fuel has been accumulated in South Korea, and it will be more than 30,000 metric tons, three times the present storage capacity, by the end of 2040 [1]. The Advanced Spent Fuel Conditioning Process (ACP) was proposed to address these challenges through the development of spent fuel conditioning technology coupled with an advanced safeguards system that will meet international safeguards goals. The goals of the ACP are to recover more than 99% of the actinides in a proliferation resistant manner and to decrease the volume and heat load from fission products. The electrolytic reduction (ER) technology developed recently by the Korea Atomic Energy Research Institute (KAERI) is expected to be a more economic, efficient, and proliferation resistant concept for the conditioning of spent fuel. With this technology, a significant reduction of the volume and heat load of spent fuel is expected, which will decrease the burden of final disposal in terms of disposal size, safety, and cost.

A source term is an estimate of the amount and radiological/chemical form of materials in the process from a specific source over a certain period of time. It is particularly important for safeguards purpose, because it refers to the first part of a calculation to estimate the contents and composition of target material that needs to be controlled. In this study, source term analyses of the ACP materials were performed to support the development of an effective non-destructive material accounting system based on coincidence neutron measurements. The goals were to produce gamma, (α,n), and

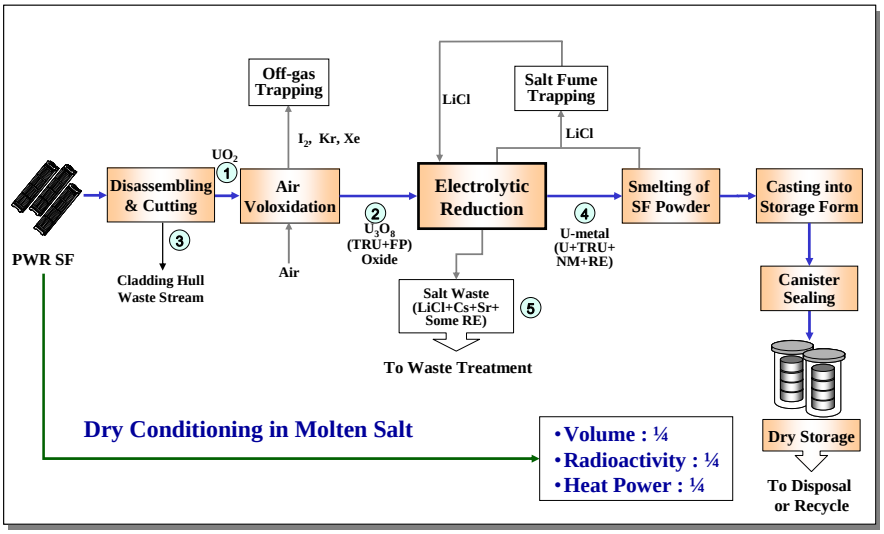

Figure 1: Material Flow of Advanced Spent Fuel Conditioning Process.

spontaneous fission neutron spectra, and to investigate fission multiplicity distributions, which may be variable across the samples in the process. The calculations were performed using ORIGEN-S of SCALE 5 for the reference spent fuel and material balance for the ACP. Lacking specific data for the ACP facility, features such as material flow pattern, lithium reduction rate, and recovery yields of elements in the products were assumed based on the conceptual design [2]. Many assumptions necessary to estimate the physical properties of the materials were also made. Based on these source term data, Monte Carlo calculations were performed to investigate the variable neutron yields of the ACP materials.

2. Material Flow and Mass Balance

The ACP technology has being developed based on the pyrochemical process that was designed in the 1960s and 1970s [3]. The reference concept consists of six major sub-processes as illustrated in Figure 1. In the reference lithium reduction process, the oxide fuel elements are chopped and voloxidized, and the resultant oxide powder is loaded into a porous magnesia basket. The basket is

		Feed UO_2	U_3O_8	Hull Waste	U-Metal	Salt Waste
Actinides		1.000	0.998	0.002	0.988	0.010
VFP		0.080	-	-	-	-
Fission Products	Se	1.000	0.998	0.002	-	0.998
	Rb	1.000	0.998	0.002	0.230	0.768
	Sr	1.000	0.998	0.002	0.220	0.778
	Te	1.000	0.998	0.002	0.230	0.768
	Cs	1.000	0.998	0.002	0.220	0.778
	Ba	1.000	0.998	0.002	0.220	0.778
	Dy	1.000	0.998	0.002	-	0.998
	others	1.000	0.998	0.002	0.998	-

Table 1: Mass Balance of the ACP Materials (20 kg HM/batch basis).

	Elements	Weight (g)	Fraction	Elements	Weight (g)	Fraction
Actinides	U	1.887E+04	9.436E-01	Pu	2.136E+02	1.068E-02
	Np	1.245E+01	6.226E-04	Am	1.789E+01	8.946E-04
	Cm	1.177E+00	5.886E-05			
VFP	I	5.867E+00	2.934E-04	Br	5.071E-01	2.536E-05
	Xe	1.389E+02	6.946E-03	Kr	8.879E+00	4.440E-04
FP	Se	1.379E+00	6.896E-05	Sb	2.365E-01	1.183E-05
	Rb	8.639E+00	4.320E-04	Te	1.215E+01	6.076E-04
	Sr	1.915E+01	9.576E-04	Cs	6.347E+01	3.174E-03
	Y	1.150E+01	5.751E-04	Ba	4.635E+01	2.318E-03
	Zr	8.696E+01	4.348E-03	La	3.121E+01	1.561E-03
	mo	8.582E+01	4.291E-03	Ce	6.282E+01	3.141E-03
	Tc	1.981E+01	9.906E-04	Pr	2.874E+01	1.437E-03
	Ru	5.864E+01	2.932E-03	Nd	1.041E+02	5.206E-03
	Rh	1.182E+01	5.911E-04	Pm	2.840E-01	1.420E-05
	Pd	3.954E+01	1.977E-03	Sm	2.190E+01	1.095E-03
	Ag	2.340E+00	1.170E-04	Eu	3.646E+00	1.823E-04
	Cd	2.867E+00	1.434E-04	Gd	3.769E+00	1.885E-04
	In	3.133E-02	1.567E-06	Tb	6.812E-02	3.406E-06
	Sn	1.396E+00	6.981E-05	Dy	3.456E-02	1.728E-06

Table 2: Material Composition of Reference PWR Spent Fuel (20 kg HM/batch basis).

charged into a reduction vessel in which the spent fuel is reduced with lithium dissolved in molten LiCl at 650 ^{o}C. Lithium and LiCl react with the spent fuel components to produce metal powders, chlorides, complex oxides etc. Some fission products with high heat load, such as cesium and strontium, are dissolved in lithium chloride molten salt, and separated from the U-metal product. The U-metal product after smelting and casting process will be held in an interim storage facility until its ultimate disposition is decided.

Recently, a modified concept of Li reduction was proposed by KAERI to simplify the technology and to increase the proliferation resistance of the process. In this electrolytic reduction (ER) process, the lithium recovery (electro-winning) step is conducted at the uranium oxide cathode simultaneously with the reduction of oxide fuel into a metallic form. Consequently, the lithium recovery process is no longer needed in this concept, and the possibility of separating actinides is inherently ruled out.

As shown in Table 1, a simple mass balance for the main unit process of the ACP was designed based on the experimental results for the chemical behaviors of the spent fuel components in LiCl [4]. The process materials in the table represent key products from each step in Figure 1 that is needed to be accounted for material balance. The following design criteria were assumed for the mass balance:

- Korean Standard PWR spent fuel with minimum 10yrs of cooling is introduced into the process. About 0.2% of initial feed material could be remaining in the cladding tube.
- Voloxidation is carried out under the air atmosphere at 500 ^{o}C to produce the oxidized spent fuel powders (U_3O_8). All the volatile fission products are assumed to be removed in the disassembling and the voloxidation processes.
- 15% excess lithium is put into the metallization process, and the final concentration of Li_2O in the LiCl salt is 3.2wt%. The LiCl+Li_2O salt can be recycled for 5 batches on a 20kgHM/batch basis.
- In case of actinides, 99% of U_3O_8 powder is reduced as a metallic form with consistent isotopic composition and the final recover rate of metallic actinides is 98.8% of feed material.

3. Source Term Analysis

The material compositions as well as radiation characteristics of the process material in ACP were estimated by ORIGEN-S based on the mass balance data in Table 1. The reference spent fuel was the Korean YoungGwang 17×17 PWR spent fuel with 3.5 wt% enrichment, 10-year cooling time, and

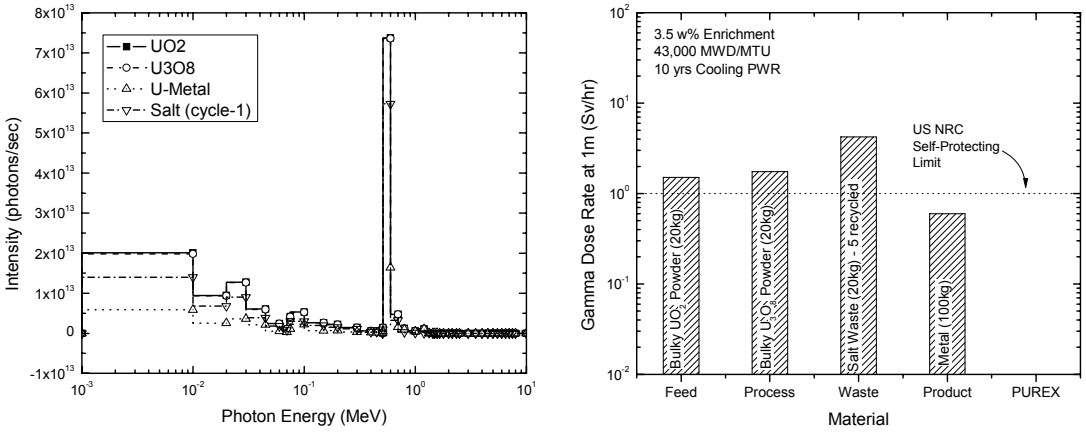

Figure 2: Gamma Spectrum (left) and Dose Rate (right) of ACP Materials (20kgHM basis).

43 GWD/MTU burn-up with 37.5 MW/MTU average power [5]. The composition of spent fuel was calculated as shown in Table 2.

3.1. Energy Distributions

Figure 2 shows the gamma radiation spectra of the process materials calculated with 44-group ENDF-V. As shown in this figure, it was found that the major contribution to gamma activity in ACP material is Cs-137 (662 keV) and the removal rate of this isotope characterize the gamma source term. Based on this data, it was also estimated that the dose rate at 1m distance from ACP materials exceed 1 Sv/hr except U-metal that is approximately 0.15 Sv/hr [2].

Neutron coincidence counting has been a valuable tool for assaying plutonium bearing materials that are pure or whose matrix constituents are well-known even in spent nuclear fuel. The multiplicity analysis equations are derived on the assumptions that all neutrons are detected with the same efficiency. However, the detector efficiency is energy dependent, so neutron counters are designed to have the detection efficiency as independent of energy as reasonably possible. In this regards, the neutron energy distribution of sample material is valuable information for the optimal design of neutron counters.

The neutron spectra of ACP materials derived by ORIGEN-S are shown in Figure 3. As it can be seen in Figure 3, it was verified that there is no significant difference in total neutron spectrum between feed

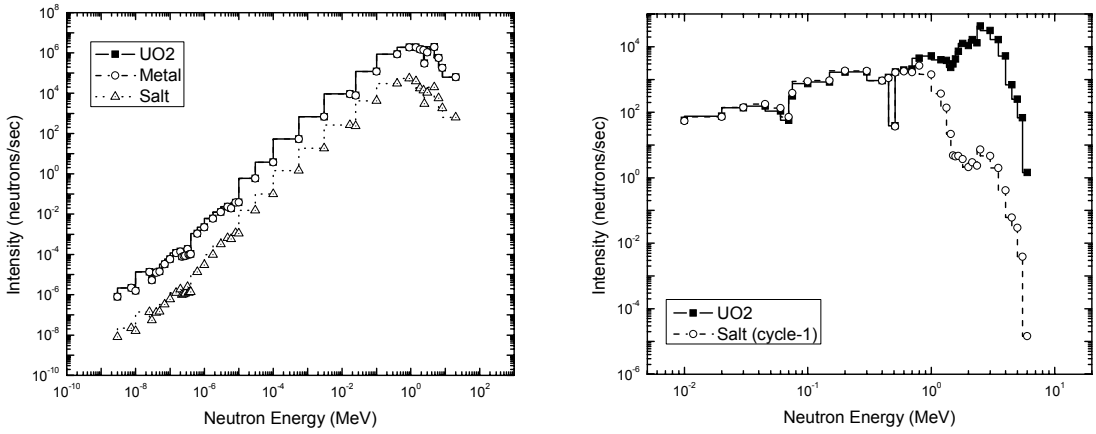

Figure 3: Total Neutron (left) and (α,n) Neutron (right) Spectrum of ACP Materials (20kgHM basis).

material (UO_2) and products (U-metal) because there is no process for isotopic changes of actinides in ACP. It also could be seen that the major energy distributions of neutron in ACP materials are from 0.1 to 10 MeV, and which means the NDA system for ACP should be designed to have constant response in this energy range.

3.2. (α,n) Neutron Rate

While the major target material for (α,n) reaction in feed material of ACP is oxygen, the salt waste has a lot of lithium and chlorine as a target element for the reaction as shown in Table 3. The ratio of (α,n) to spontaneous fission neutrons (the quantity "α") is an important factor to be considered to design an optimum NDA system. Passive neutron counting usually shows significantly lower accuracy for such samples and assay precision is degraded to the point that long count times are required for acceptable results. Therefore, one of the specific goals for neutron counter design is to minimize the effects of variations in the sample's emitted neutron energy spectrum due to (α,n) reactions or sample moderators [6].

Based on the batch mode calculations with blending option of ORIGEN-S code, the quantity "α" of the samples in ACP were derived as summarized in Table 3. The results were calculated from (α,n) spectra of process materials as shown in Figure 3. According to these results, it was known that the feed materials have similar (α,n) neutron spectra and "α" value. However, salt waste material shows different spectrum in energy range of 1~5 MeV, and the "α" value was calculated as about 0.7. It was also found that the major contribution to the (α,n) reaction of salt waste is Li-7 by Cm-244. Even though the "α" value of salt waste is 40 times higher than feed materials, it could be considered as relatively low "α" value regarding other impure matrices of high "α" value [7]. For the ACP application, therefore, neutron coincidence counting using the known-alpha analysis method would be a reasonable approach considering relatively low range of "α" values.

3.3. Fission Multiplicity Distributions

The neutron emission multiplicity distribution P_ν, the probability that a fission will result in ν neutrons, is essential information in the development of methods of self-calibration or standardization of instrumentation [6]. Therefore, it is important to develop as accurate and complete a P_ν data base for ACP as possible in order to support the development or evaluation of empirical or theoretical calculations of neutron coincidence counting. Such calculations or correlations in turn can be used to predict neutron counting statistics or related values for samples of ACP for which experimental approaches are too difficult.

The nuclear parameters P_ν, and $<\nu>$ that are of particular safeguards interest, have been evaluated for sample materials of ACP. The values presented in Table 4 were derived from the latest coincidence capability of the MCNPX code that provides the Singles, Doubles (and Triples), and the factorial moments of fission multiplicity. As shown in Table 4, multiplicity distributions from the spontaneous fission of ACP material show quite similar values to the canonical "consensus" set of safeguards community [8]. It was also found that the induced fission multiplicities of ACP materials are slightly larger than typical U-235 case, and it's rather similar to the prompt induced fission multiplicity of Pu-239 [9].

	Bulk Density (g/cm^3)	Nominal Density (g/cm^3)	(α,n) neutron		SF neutron		α
			Intensity (n/s)	Avg. Energy (MeV)	Intensity (n/s)	Avg. Energy (MeV)	
UO_2	2.0 ~ 5.0	5.0	2.04E+05	2.40	1.18E+07	2.10	0.0172
U_3O_8	1.5 ~ 4.5	4.33	2.41E+05	2.40	1.18E+07	2.10	0.0204
Hull	2.0 ~ 5.0	5.0	4.08E+02	2.40	2.36E+04	2.10	0.0172
U-Metal	18.0 ~ 19.0	18.0	6.39E-02	4.95	1.17E+07	2.10	0.0000
Salt	1.5 ~ 2.5	2.0	8.24E+04	0.66	1.18E+05	2.10	0.6966

Table 3: Neutron Characteristics of ACP Materials (20 kg HM/batch basis).

Probability Distribution	Spontaneous Fission		Induced Fission				
	ACP	Cm-244[8]	UO_2	U_3O_8	U-Metal	Salt	Pu-239[9]
P(0)	0.015	0.015005	0.037	0.037	0.037	0.045	0.011
P(1)	0.117	0.116172	0.124	0.123	0.121	0.123	0.101
P(2)	0.300	0.299842	0.259	0.259	0.258	0.266	0.275
P(3)	0.332	0.333161	0.303	0.303	0.303	0.266	0.324
P(4)	0.183	0.183774	0.195	0.194	0.195	0.195	0.199
P(5)	0.044	0.042978	0.066	0.068	0.069	0.097	0.083
P(6)	0.009	0.008791	0.014	0.015	0.014	0.006	0.008
P(7)	-	0.000274	0.002	0.002	0.001	-	
$<\nu>$	2.717	2.720	2.759	2.765	2.769	2.760	2.879
$<\nu(\nu-1)>$	5.928	5.939	6.500	6.541	6.554	6.610	6.773
$<\nu(\nu-1)(\nu-2)>$	10.066	10.101	12.560	12.742	12.705	12.896	12.630
$<\nu(\nu-1)>/<\nu>^2$	0.803	0.8027	0.854	0.856	0.855	0.868	
$<\nu^2>-<\nu>^2$	1.263	1.26	1.647	1.661	1.654	1.754	
$<\nu^2>$	8.645	8.659	9.259	9.306	9.324	9.370	

Table 4: Neutron Multiplicity Distributions of ACP Materials.

4. Summary

In this study, neutron and gamma-ray source terms have been analyzed to provide background data for the development of effective safeguards system for ACP. Based on the source term data, neutron multiplicity distributions of various ACP sample materials were also determined with the latest coincidence capability of the MCNPX code. The results of this study would be useful for the development of optimized nuclear safeguards system that could meet IAEA safeguards criteria.

5. Acknowledgements

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6. References

[1] Li, T.K. et al., *A Study on the Safeguardability of Advanced Spent Fuel Conditioning Process*; LA-UR-04-4187; Los Alamos National Laboratory; 2004.
[2] Li, T.K. et al., *Safeguardability of Advanced Spent Fuel Conditioning Process*; Proceedings of 45th INMM Annual Meeting; Institute of Nuclear Materials Management; 2004.
[3] Karell, E.J., Pierce, R.D., Mulcahey, T.P., *Treatment of Oxide Spent Nuclear Fuel Using the Lithium Reduction Process*; ANL/CMT/CP-89562; Argonne National Laboratory; 1996.
[4] Kim, Ik-Soo et al., *Characteristics of Reduced Metal from Spent Oxide Fuel by Lithium*; J. KNS; Vol. 35, No. 4; 2003; p 309-317
[5] Gauld, I.C., Hermann, O.W., Westfall, R.M., *ORIGEN-S: Scale System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Build up and Decay, and Associated Radiation Source Terms*; NUREG/CR-0200 (Rev. 7); Vol. 2, Sec. F7; USNRC; 2002.

[6] Reilly, D. et al., *Passive Nondestructive Assay of Nuclear Materials*; LA-UR-90-732; Los Alamos National Laboratory; 1990.
[7] Geist, W.H. et al., *The Russian Epithermal Neutron Multiplicity Counter (RENMEC)*; LA-UR-03-4077; Los Alamos National Laboratory; 2003.
[8] Holden, N.E., Zucker, M.S., *Neutron Multiplicities for the Transplutonium Nuclides*; BNL-36467; Brookhaven National Laboratory; 1985.
[9] Boldeman, J.W., Hines, M.G., *Prompt Neutron Emission Probabilities Following Spontaneous and Thermal Neutron Fission*; Nucl. Sci. Eng.; Vol. 91, No. 114; 1985.

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Neutron Detection for Process Monitoring of Stores

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Abstract:

Neutron monitors are extensively used in safeguards to detect the passage of nuclear material. The principal advantage over camera surveillance is that the neutron signal indicates whether or not the item contains nuclear material. In addition to determining individual item movements, the counting rate from neutron detectors can give a measure of the total nuclear material inventory in storage or process areas. This paper shows how neutron monitors that are intended for detection of movements can be used to monitor nuclear material inventory. The background counting rate in the monitors has been calculated as a function of the number of items stored. This counting rate is sufficient to draw some conclusions about the total number of items in the store. Sufficiently sensitive neutron monitors would also record any internal movements in the store and thus would be able to confirm if the store was being operated as declared.
In standard safeguards applications, cameras are often used to ensure that neutron monitors are still effective by ensuring that no shielding material has been placed around them. Using cameras for this purpose means that the neutron monitoring system cannot be considered a single layer of containment and surveillance by itself because it needs the camera system to ensure that it is still operational. We describe how the counting rate of the neutron monitor itself can be used to determine if neutron shielding has been applied. In this way, neutron monitors can be considered as a stand-alone layer for containment and surveillance purposes.
Numerical examples of the counting rates in a conceptual plutonium storage area are presented and the sensitivity of the system to removal of containers is estimated.

Keywords: Nuclear material stores, dual C/S, Monte Carlo, neutron transport, process monitoring

1. Introduction

Neutrons are emitted in large quantities from irradiated fuel and plutonium. They are penetrating and difficult to shield. For these reasons, many storage facilities containing irradiated fuel or plutonium use neutron monitoring to monitor declared (and discover undeclared) movements of nuclear materials. The primary purpose of the monitors is to indicate that neutron-emitting material has passed by the monitor. It is the intention of this paper to demonstrate that under some circumstances it is possible to use the counting rates to give an indication of the total amount of material in the store. In addition, the counting rates themselves can indicate an attempt by the operator to add shielding to the monitors to conceal nuclear material movements.

2. Monitoring of store contents

We have simulated a PuO_2 storage room using the Monte Carlo (MCNPX) code [1] to estimate the neutron monitor counting rates. The configuration for the simulation was a large storage vault with concrete walls, floor, and ceiling. The room dimensions are approximately 20m long x 10m wide x 6m high with a material portal at one end and a separate personnel door near the wall, as illustrated in fig. 1.
The cans containing PuO_2 are assumed to be stored behind a concrete shield that is ~ 0.5 m thick to reduce the dose to personnel in the vault. A typical can is assumed to contain 5 kg of reactor grade

plutonium and the neutron background in the room is primarily a function of how many cans are in storage. The cans are brought into the vault through the entry portal and personnel enter through the smaller door shown in Fig. 1. The neutron monitors are set up in two positions near the material entry to indicate the direction of motion and a third monitor is near the personnel door. A can that is removed from its storage shield will raise the signal level in all three detectors, and a can that passes through a portal will provide a peak in the sensor located near the portal.

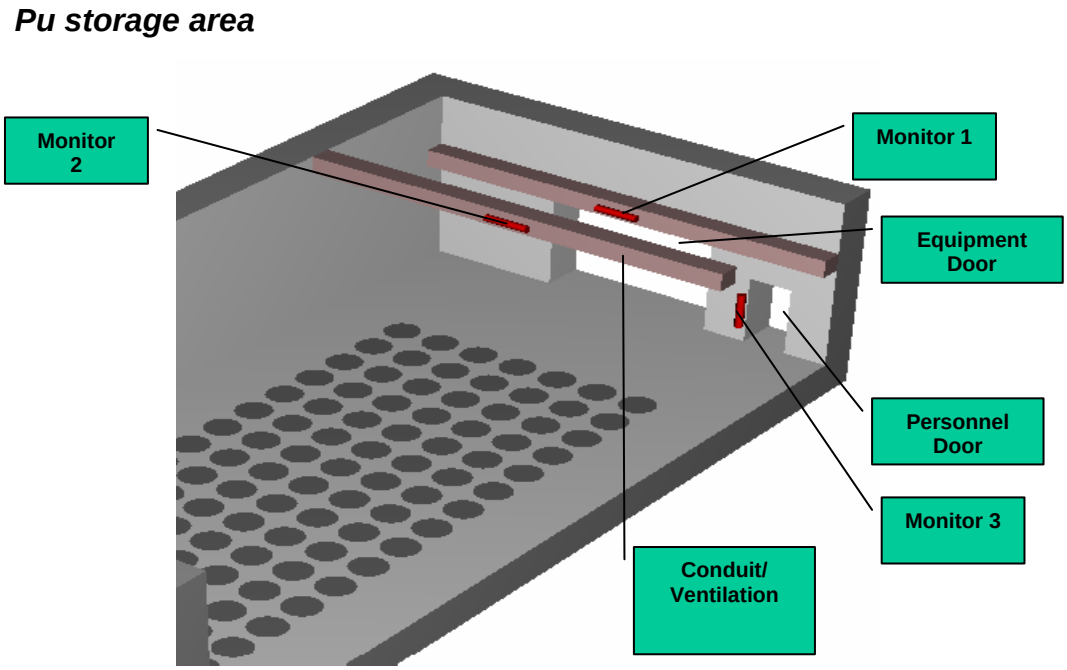

Figure 1: Simulated PuO_2 storage room used for the MCNPX calculation of signal and background rates.

Despite the large amount of shielding, each can makes a significant contribution to the counting rates in the monitors. The counting rate in the neutron monitors caused by 1 can that contains 5 kg of PuO2 in a storage position is roughly 0.05 counts/sec/can. This calculation is rough because it depends on the can position and the concrete thickness. The background count rate in the monitors would show a clear pattern as the store was filled. The rate as a function of time would depend on how the store was filled. Figure 2 shows the results of Monte Carlo calculations for two extreme cases, when the store is filling from the end far from the monitor and when it is filling from the near end.

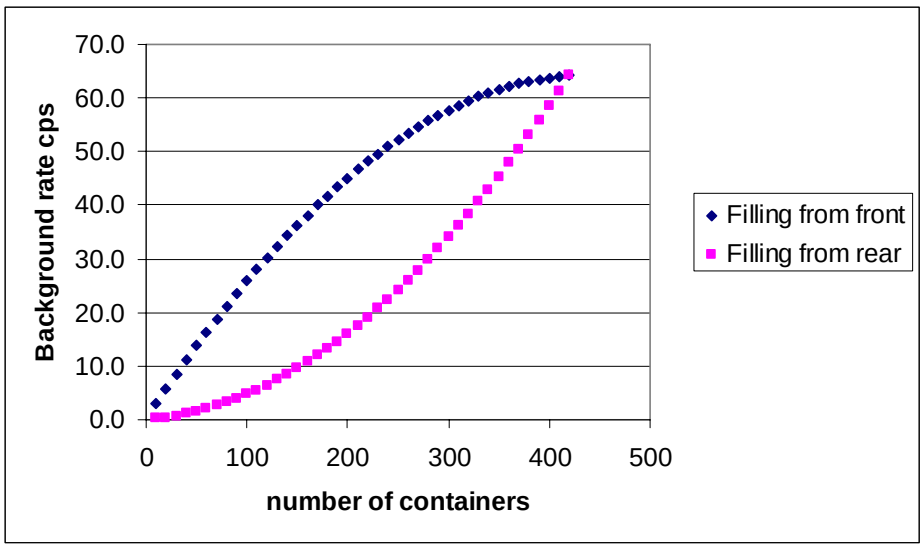

Figure 2: Background Counting Rates as store fills

It is interesting to consider how sensitive this measurement would be to the (undeclared) removal of a can. The counting rate change depends on the position from which the can is removed. The detection capability depends on the counting statistics of the background rate. If we assume that we can accumulate background counts for a period of 8 hours (i.e. 1 shift involves no movement of material) we obtain the results shown in figure 3.

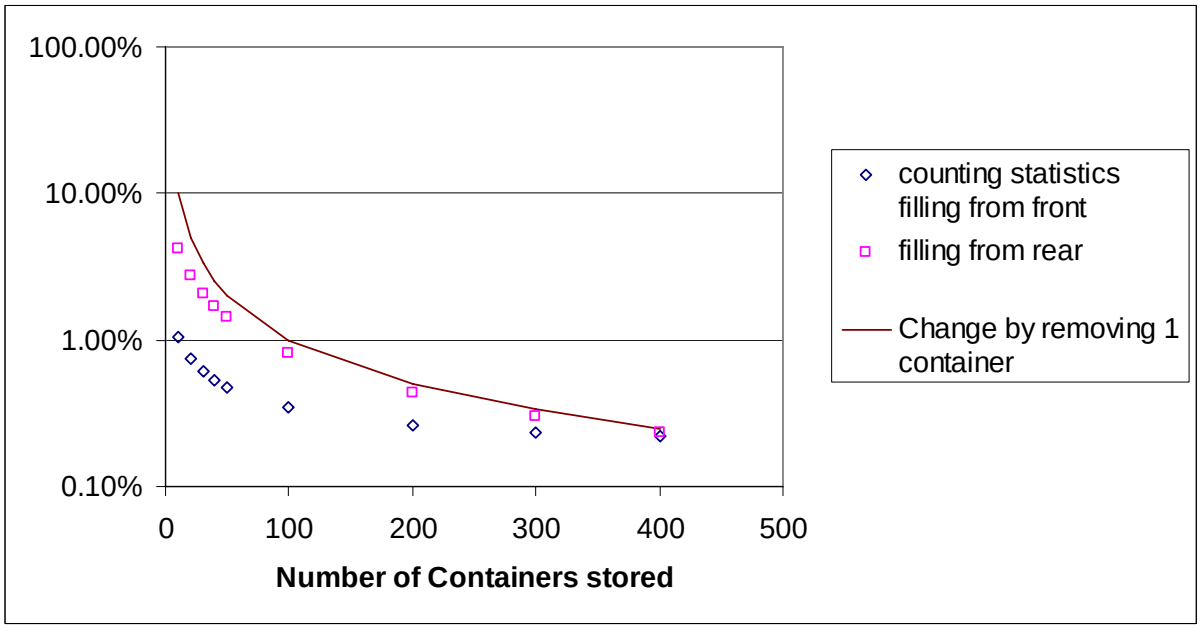

Figure 3: Count rate change by removing one container compare to the statistical error

The figure compares the percentage change in count rate from a can removal to the percentage statistical uncertainty of the background determination. The solid curve represents the change in counting rate caused by the removal of one container. This gets smaller as the store fills. The points represent 3 sigma of counting statistics. The percentage statistical uncertainty is less when the background is larger and it also decreases as the store fills. What this rough calculation tells us is that when the container removal gives a 0.5-1% change, there is a very good possibility of detecting this through a change in the background counting rate. The situation worsens as the store fills simply because the percentage change caused the removal of one container becomes very small. In order to

make a good estimate of the detection limits, a precise model of the store in question would be needed. In a real store where data is collected continuously, the background would create a kind of calibration curve of the effect of each container as the store is filled without the need for modelling.

Potential objections to the use of this method come from the use of additional neutron sources and/or neutron shielding by the operator. The second point, under normal circumstances, could be covered by the use of video surveillance or in case of surveillance failure by the technique discussed in the second part of this paper. While additional neutron sources, such as ^{252}Cf could be easily used to change the count rate in one detector it would be difficult to precisely compensate for missing material. It would be extremely difficult to compensate for nuclear material removal precisely in a number of neutron monitors simultaneously.

This method of monitoring the store contents, which is simply additional analysis of data that is automatically collected, could be a measure used to give additional confidence that the plant is operating as declared.

3. Detection of Applied Shielding

For current applications of neutron monitoring, the surveillance system includes cameras to monitor the neutron sensor location to show the absence of neutron shielding. The successful application of both the camera system and the neutron monitor are required to make use of the radiation sensor. Thus, if the video system fails, the neutron results do not provide assurance that there has been no undeclared transfer of the nuclear material. This paper presents a technique for a neutron sensor to monitor for the application of neutron shielding that might be used to cover the sensor. The basis of the technique is that the continuous operation of the monitor establishes a background rate from external source neutrons. If shielding is applied around the sensor, the background rate drops to a level that can be identified in the data review software. Typical sources of the background continuum are cosmic-ray spallation source neutrons and stored nuclear fuel such as MOX and spent fuel. In most cases, two or more neutron sensors are positioned in the room to establish direction of motion and to be near portals of entry or exit. The counting rate of the monitors will indicate the presence and direction of movement of the containers.

In most nuclear facilities, there are neutron backgrounds coming from both cosmic-ray source neutrons and stored nuclear materials. The stored materials include PuO_2, MOX, and spent fuel. The neutrons are born with high energy from spallation reactions, spontaneous fission, and (alpha,n) reactions. Before reaching the neutron sensor, the neutrons have travelled large distances and undergone many scattering reactions. Thus, the direction of the neutron's travel is changed many times and it is necessary to completely surround the sensor with shielding to prevent the monitor from seeing an undeclared transfer. However, this shielding also reduces the room background rate so that it can be flagged in the data review software.

To demonstrate the concept, we used the same model as before, as shown in figure 1.

In general, it is a good idea to position the sensors adjacent to building structural components, such as metal conduit runs, so that neutron shielding is blocked from one direction. Figure 1 shows the two entry sensors attached to building conduits. Also, the neutron sensor positioned near the doorway in Fig. 1 cannot be completely shielded without physically blocking the door's passageway.

A typical neutron monitor contains two He-3 tubes and amplifiers are used to provide redundancy in case of failure. The data is collected continuously with 20-s readout intervals. Normal time intervals between sample movements are many minutes, so the background continuum is determined over hour-type intervals. All of the neutron monitors, when unshielded, are capable of detecting any can that is removed from the storage cells. The failure modes of the neutron monitoring systems are 1) the system itself fails 2) the operator adds shielding around a can, canister, or the monitors themselves so as to prevent the detection of material movement.

The background count rate from cosmic-rays was estimated to be ~ 0.3–10 depending on the amount of concrete overhead. The measured count rate would be the sum of the cosmic-ray neutrons and the shielded cans of PuO_2. When a can is moved passed the monitor, the count rate increases by over

three orders of magnitude. Shielding around the neutron monitor would have to be over 300 mm thick to hide the transfer of the can. The resulting shield weight would be ~ 500 kg. With 10 cans in the store, the background counting rate would be ~ 0.8 cps. Looking at the statistical accuracy obtainable from these counting rates, we require a measurement time of around 40 seconds to be able to clearly distinguish the shielded background condition. If however, the counting rate did turn out to be too low, it would be possible to place (and seal) a small neutron source in the neighbourhood of the monitors to artificially increase the counting rate.

Figure 3 shows a simulation of the counting rate in one monitor with a low background rate. The values plotted are the counts recorded in 20-second intervals. The peak that is shown is due to the normal movement through the portal of a shielded PuO2 can. The peak from an unshielded can would correspond to over 10,000 counts on the same scale. The application of the shielding is clearly indicated in the simulated data where the shielding was applied for about 10 min.

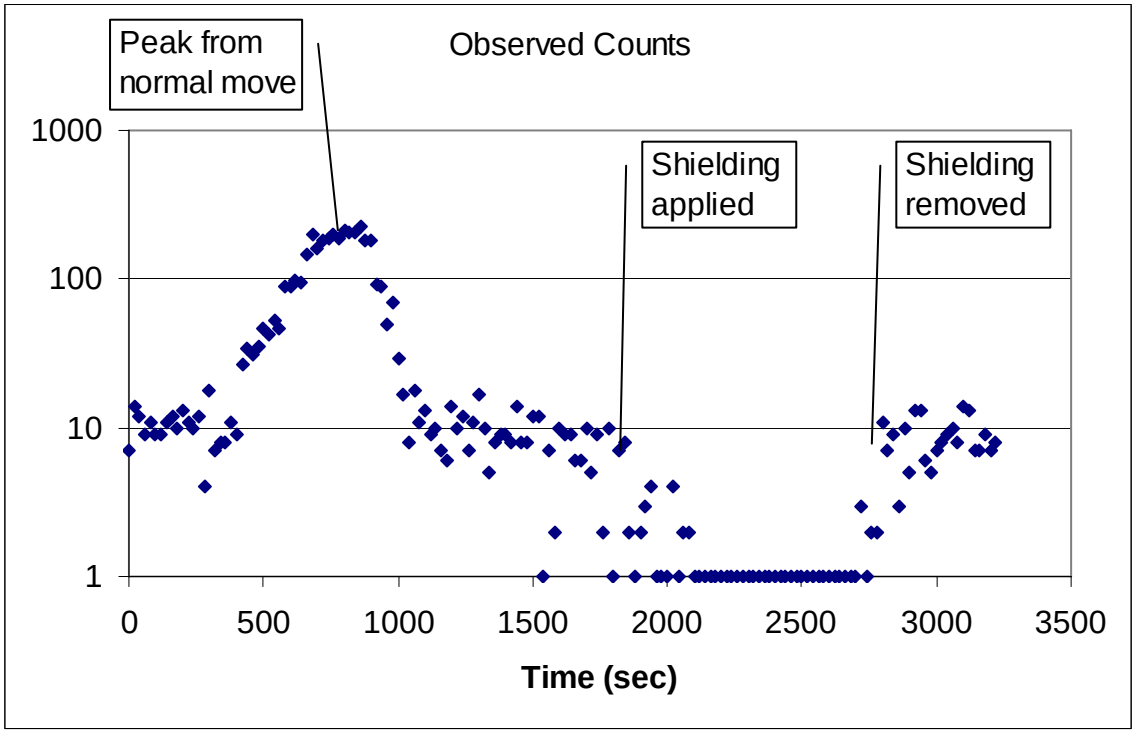

Figure 3: Counts recorded per 20 seconds

This analysis might appear to be time-consuming because the data would need to be checked whether there were material movements or not. However it would be relatively simple to automate the task, but even this may not be necessary because it is important to remember that this analysis would only be required during the period of surveillance failure, which should make it a very unusual occurrence.

In general, the monitors should be placed close to the actual path of movement of material so that extra shielding cannot be interposed. Other useful locations are near to structural parts of the building that prevent the addition of shielding on one side—for example steel beams and conduits.

The video system is capable of seeing the activity in the store. However the video system alone cannot tell if the transfer can and cart is full or empty. In addition, the removal of a can might be difficult to determine by the video system alone. For example if a large number of people entered the store carrying tools and large equipment for maintenance, a can could be concealed when they left. The failure modes of the video system are 1) the system itself fails, 2) the lights fail.

The dual C/S status of the plutonium storage areas is possible using video surveillance and neutron monitors. In most cases, there are more than one neutron monitor per storage area. Some diversion

scenarios may not be easy to detect with an individual system and therefore the video and neutron data acquisition systems should be designed to be tolerant to single failures. When multiple neutron sensors are used in the same storage area, it would be necessary to apply massive amounts of shielding to all monitors at the same time and this would be easy to identify in the data review software.

4. Conclusion

The data acquired from neutron monitors may allow them to play a greater role in the safeguarding of neutron stores than detecting the passage of items.

The background counting rate can, in some circumstances, indicate the total amount of material in a store and detect undeclared removal of material. The results available from such monitoring will detect on the precise arrangement of the monitors and the store. Neutron monitors at both ends of such a store would give better performance in terms of both rates and uniformity.

By analysing the data from the detectors during periods of video surveillance failure, it should be possible to detect the addition of shielding around the detector. This means that neutron monitors alone could be considered an independent layer of containment and surveillance.

5. References

1. J. Hendricks et al. “MCNPX Extensions Version 2.5.0” Los Alamos National Laboratory Report LA-UR-04-0570

Identification and quantitative assay of radioisotopic neutron sources by neutron and gamma measurements

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Abstract

Time-correlated neutron emission from spontaneous fission, (n,fission) and (n,2n) reactions was investigated based on neutron coincidence experiments for various neutron sources. According to a simple model, the neutrons were assumed to be randomly emitted in a spherical source, and induce neutron-producing secondary reactions in the material. Calculation shows that the neutron coincidence count rate, R, depends on neutron energy and the type and amount of isotopes in the source material. The total count rate, T, depends only on the amount of the material in a source of a given type. High resolution gamma spectrometry is a powerful method for identification of the nuclide composition and also for determining the activity and thereby the transuranium content of a source, correcting for self-absorption of the prominent gamma-ray intensities. The type of the source can be characterized and identified as well, as an independent method, by neutron measurements only, based on the ratio R/T (in the case of spontaneous fissioning neutron sources, R/T is constant for a type of material). The transuranium contents can also be estimated by using a single neutron measurement, in a reasonable agreement with the model calculations. Contents of ^{252}Cf, ^{244}Cm, AmBe, AmLi, ^{238}PuBe, and ^{239}PuBe neutron sources were assessed by this method.

Keywords: NDA; neutron sources; coincidence measurements; gamma spectrometry

1. Introduction

In the last years, non-destructive neutron and gamma methods were developed for assaying Pu-Be neutron sources. As a first step, the isotopic composition of intense neutron sources (10^5-10^7 neutron/s) was determined by gamma-spectrometry using a planar HP-Germanium detector. By combining it with the total neutron output, their Pu content was estimated. The results were reported in Refs [1, 2]. On the other hand, Pu can be assayed using gamma-spectrometry only, by the so-called infinite energy method, with an uncertainty of about 20% [3]. Recently, the Pu-content of Pu-Be neutron sources was estimated in a far-field assay based on intensity measurements of the 375 and 413 keV gamma lines of ^{239}Pu [4]. This method has been developed to improve the accuracy by correcting for self-absorption instead of an extrapolation to the infinite energy.

Upon developing these methods, another approach, as a feasibility study, was attempted, using a correlation of the ratio R/T (R is the real coincidence count rate and T is the total count rate from neutron measurements) with the Pu-content determined by the combined or by the pure gamma-spectrometric method [1, 2]. This correlation seems to be insensitive to isotopic composition and could be used to estimate the Pu-content, and can be called R/T-method. The ratio R/T was first used in correcting for the multiplication upon measuring the ^{240}Pu$_{eff}$ content of Pu samples by neutron coincidence measurements [5].

In this paper we report on a simple model calculation, in which the random neutrons are assumed to be created in the centre of the spherical neutron source, and induce neutron-producing secondary neutrons around this point, in order to explore the principle and performance of the R/T method. This model was used for calculating the increase in neutron emission rate due to Be(n,n), Be(n,2n) and

Pu(n,f) reactions in Pu-Be neutron sources [6, 7]. We adopt the above type calculation and also develop a method for calculating coincidence count rate. Moreover, this method can be used to determine the transuranium content of various radioisotopic neutron sources.

The calculation shows that the relationship between the R/T and T can be applied to identify the type of an unknown neutron source. The transuranium content of the source can be assessed by the combined neutron-gamma method or the independent method of pure gamma spectrometry. On the basis of a correlation between R/T and the transuranium content, R/T can be calibrated in terms of transuranium mass. Better accuracy can be achieved by calibrating the method using calorimetry, see the companion paper presented in this symposium [8].

2. Determination of the transuranium content in radioisotopic neutron sources by the coincidence technique

2.1. Determination of the Pu content in ^{239}Pu-Be neutron sources

Plutonium isotopes emit α-particles with various energies and specific yields. Due to the $^{9}Be(\alpha,n)^{12}C$ reaction, the mixed material of Pu and Be becomes a random neutron-source. The neutrons interact with the material in the source by the following processes [6]: i) elastic scattering on Be nuclei; ii) Be(n,2n) and iii) Pu(n,f) reaction, inducing time-correlated secondary neutrons in the energy region of 0-2 MeV. The calculation and experiments [6] show that although the elastic scattering takes place with high probability (0-29 %) and change the direction of neutrons, it does not affect neutron energy spectra. The probability of Be(n,2n) and Pu (n,f) reaction is about (0-5)% and (0-2)%, respectively. We assume that the Be(n,n) reaction does not increase the secondary neutron output and the Be(n,2n) and Pu(n,f) reactions are independent to each other. Further assumptions are that the composition of the source material is an intermetallic compound $PuBe_{13}$, with a density ρ=3.7 g/cm^3. The probabilities p_1 and p_2 of the secondary-neutron producing Pu(n,f) and Be(n,2n) reactions, respectively, induced by one primary random neutron, are

$$p_i = Dn_i\overline{\sigma}_i/2 \ ; \qquad p = \frac{D}{2}\sum n_i\overline{\sigma}_i \tag{1}$$

where D is the diameter of the source, n_i is the atom density of Pu or Be in the source (atom/cm^3), and $\overline{\sigma}_i$ is the respective cross section [9] averaged over neutron energies, numerically calculated as:

$$\overline{\sigma}_i = \int f(E)\sigma_i(E)dE \ . \tag{2}$$

Here $f(E)$ is the spectral density at neutron energy E, and the index i runs over 1 through 2.

The increment Δt of the total count rate due to secondary neutrons and the detected count rate of double coincidences (doubles) r, induced by a single random neutron, can be written as

$$\Delta t = \sum \Delta t_i = \sum p_i C_{1i}\overline{\nu}_{1i} \tag{3}$$

and

$$r = \sum r_i = \sum p_i C_{2i}\overline{\nu}_{2i}, \tag{4}$$

respectively, where $\overline{\nu}_{1i}$ is the average of the prompt neutron multiplicities of the two induced reactions, $\overline{\nu}_{2i}$ is the second order factorial moment of the multiplicity distribution of prompt induced neutrons, C_{1i} and C_{2i} are instrumental constants depending on the parameters of the experimental set-up; ε_i is the absolute detector efficiency, P is pre-delay, G is the coincidence gate length, and τ is detector die-away time, according to the formulae

$$C_{1i} = \varepsilon_i \tag{5}$$

$$C_{2i} = \varepsilon_i^2 e^{-P/\tau}(1 - e^{-G/\tau}) \tag{6}$$

Table 1 shows the values of the parameters used in the calculation.

Isotope	$\bar{\sigma}$ (b)	Atom density-*n* (atom/cm^3)	$\bar{\nu}_1$	$\bar{\nu}_2$
^{239}Pu	2.1	6.5 10^{21}	3.5	5.5
^{9}Be	0.4	6.5x13x 10^{21}	2	1

Table 1. Values of the parameters used in calculating ^{239}PuBe neutron sources

The values $\bar{\nu}_{11}$, $\bar{\nu}_{21}$ of ^{239}Pu were taken to be higher than those tabulated for thermal neutrons, because they increase with neutron energy. Data in Ref. [9] and correlation between $\bar{\nu}_{11}$ and $\bar{\nu}_{21}$ from systematics [10] were taken into consideration.

Instead of source diameter D we introduce source mass M_{PuBe} and total Pu mass M_{Pu}, according to

$$M_{PuBe} = \frac{4\pi D^3 \rho}{3*8}, \text{ i. e. } D = \left(\frac{6M_{PuBe}}{\pi\rho} \right)^{1/3} \tag{7}$$

and, taking into account that M_{PuBe} =(239+9∗13)/239M_{Pu}, finally we obtain

$$D = 0.92 M_{Pu}^{1/3}. \tag{8}$$

In this way the probabilities p_i and hence Δt and r can be expressed as a function of Pu mass. For ε_i of an experimental setup we used one single ε value as a first approximation. Figs 1, 2, and 3 display these quantities for a single random neutron as a function of Pu-content.

The calculations for the 0.01-120 g range of M_{Pu} show that:

- The individual probabilities are 0-7% for the Be(n,2n) and 0-3% for the Pu(n,f) reaction;
- About 60 % of multiplication comes from the (n,2n) reaction and about 40 % of that from the Pu(n,f) reaction;
- However, only 30 % of doubles is due to the (n,2n) reaction, while 70 % to Pu(n,f);

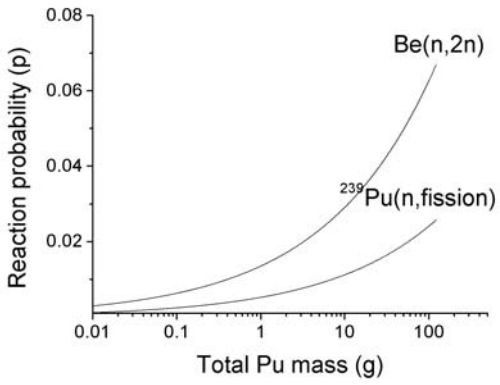

Fig. 1. The reaction probability *p* as a function of total Pu mass

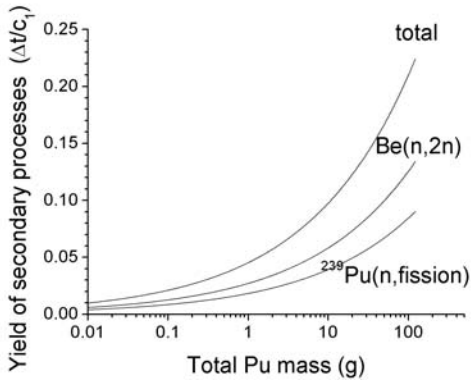

Fig. 2. Neutron yield of secondary reactions

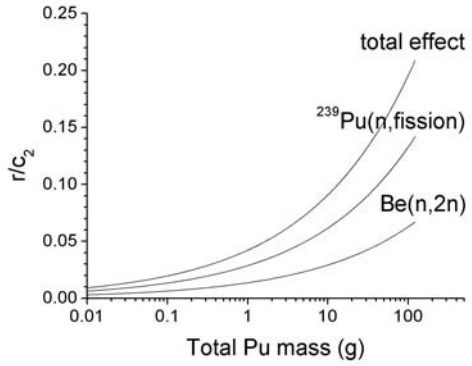

Fig. 3. The yield of double coincidences

By denoting primary and total neutron emission rates by N_0 and N, respectively, the total count rate $T=\varepsilon N=N_0(\varepsilon+\Delta t-\varepsilon p)$ and the doubles count rate $R=N_0 r$ can be determined from the measurement. Then r and R/T can be expressed as functions of Pu mass as

$$r = 0.46 M_{Pu}^{1/3} \sum C_{2i} n_i \overline{\sigma}_i \overline{\nu}_{2i}, \tag{9}$$

and

$$R/T = r\left(\frac{1}{\varepsilon(1-p)+\Delta t}\right). \tag{10}$$

Fig. 4 shows the measured values of the ratio R/T as a function of the total Pu-content for three detector-types of efficiencies 2.8, 8.8, and 28%. The former two are home-made [1, 2], while the latter is a commercial neutron coincidence collar type JCC-13, made available to us by EC JRC Ispra, Italy. We note that strong neutron sources (above 10^7 n/s) have to be measured by low detector efficiency. However, small neutron sources should be assayed by as high detector efficiency as possible. In Fig. 4, for the detector of efficiency 8.8 %, two curves are presented; one was calculated and the other is a simple fit to the experimental points by a second order polynomial function. The experimental curve deviates from linearity toward high masses. This is due to the further multiplication of secondary neutrons, which is not taken into account in this simplified model.

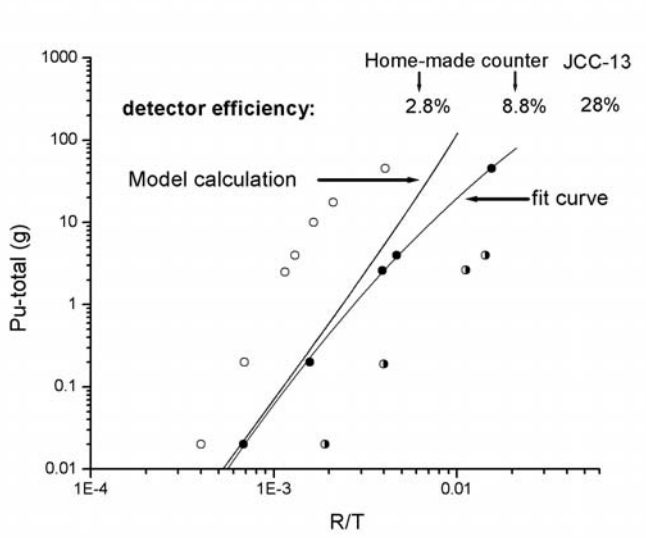

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Fig. 4. Total Pu-mass as function of the ratio R/T for different detectors

The experimental data are presented in Table 2. The actual neutron outputs are calculated from the original data given by the supplier. The Pu contents are determined by the pure gamma-spectrometric method, as discussed below.

Parameters of detectors and set-up					
Efficiency	2.8%	8.8%	28%		
Die-away (μs)	70	100	40		
Pre-delay(μs)	30	30	7		
Gate (μs)	64	128	60		
Nominal Pu content (g)	**R/T (10^{-3})**			**Actual neutron output (10^5n/s)**	**Pu-content (g)**
0.18	0.4(0.04)	0.72(0.06)	1.9	0.117	0.02(0.002)
1.8	0.7(0.20)	1.78(0.2)	4.0	1.17	0.20(0.02)
4	1.28(0.21)	4.23(0.4)	11.2	2.99	2.55(0.10)
36	1.29(0.23)	6.23(0.3)	14.3	24.7	4.0(0.15)
45	1.64(0.15)			59.8	9.5(0.5)
177	2.16(0.20)			114	19(1.5)
85	4.19(0.43)	21.7(1.1)		57.2	50(3)

Table 2. Parameters of the detectors, the neutron sources, and the measured values of the ratio R/T

2.2. Determination of transuranium content in neutron sources

In the same way, the above calculations can be performed for all neutron sources containing transuranium isotopes, in which such secondary processes take place. That is, the transuranium content of AmBe, ^{238}PuBe and AmLi neutron sources can be determined by this *R/T*-method by establishing a calibration on the basis of the correlation between *R/T* and M_{Tru} for each type neutron source. Because the values of $\overline{\sigma}$ (^{238}Pu)=2.4, $\overline{\sigma}$ (^{240}Pu)=1.7, and $\overline{\sigma}$ (^{241}AmBe)=1.4 b, are close to that of $\overline{\sigma}$ (^{239}Pu)=2.1 b in the related energy region, their calibration curve is similar to that of PuBe neutron sources. The energy spectrum of AmLi sources is shifted toward lower energies, so $\overline{\sigma}$ (^{241}AmLi)=0.2 b is smaller than that of AmBe sources by about a factor of an order of magnitude. Experimentally, the R/T value of our AmLi source is smaller than those of AmBe sources by a factor of about 6-8.

We have measured R/T values for AmBe, ^{238}PuBe, and AmLi sources as reported in Ref. [11], in reasonable agreement with the results of model calculation. In Ref. [12] a series of measured data of AmBe and AmLi neutron sources are also in general agreement with it. Because the exact experimental conditions are not known, the individual cases cannot be discussed in detail.

3. Source identification

Gamma spectrometry is a powerful tool for identifying the type of neutron sources [11, 12]. Coincidence counting can also be used for identification of the type of neutron sources. The transuranium mass M_{Tru} can be found out from measured source strength (N) and the specific neutron yield (Y) from the literature, using the relation

$$M_{Tru} = \frac{N}{Y} = \frac{T}{\varepsilon Y} \ . \qquad (11)$$

By inserting M_{Tru} from the formula (11) into (9), then the expression obtained for r into (10), *R/T* as a function of T can approximately be obtained as

$$R/T = 0.46 \frac{T^{1/3}}{\varepsilon^{4/3} Y^{1/3}} \sum C_{2i} n_i \overline{\sigma}_i \overline{\nu}_{2i} \qquad (12)$$

The curve, representing the *R/T* values for AmBe, ^{238}PuBe, and AmLi as a function of T in our experiments, splits into individual groups of data corresponding to the values of the specific neutron yield Y characterizing a type of neutron sources. It is known that in the case of spontaneous fissioning

neutron sources, *R/T* is constant for a type of material. The values of *R/T* as a function of *T* in log-log scale are shown in Fig. 5 for ^{244}Cm, ^{252}Cf, AmLi, AmBe, ^{238}PuBe, and ^{239}PuBe neutron sources.

In case of ^{239}PuBe, the curve splits into two groups, corresponding to ^{239}Pu abundances of 95% and 75-80%, because of their different specific yields. The data of Ref. [12] for the ratio R/T of AmBe and AmLi, as plotted in Fig. 6, are in agreement with our calculation again.

From the results of the model calculation and the experimental data, a procedure for identifying an unknown neutron source can be considered as follows:

- The source is to be measured by neutron coincidence technique for determining the parameters R and T, then R/T.
- Plot it in a figure similar to Fig. 6 constructed from a set of standardized neutron sources for estimating the type of the source.
- The transuranium content is determined by using a calibration between R/T and M_{Tru}, corresponding to the type of the assayed neutron source, or using formula (11) for the neutron output N and the specific yield Y.

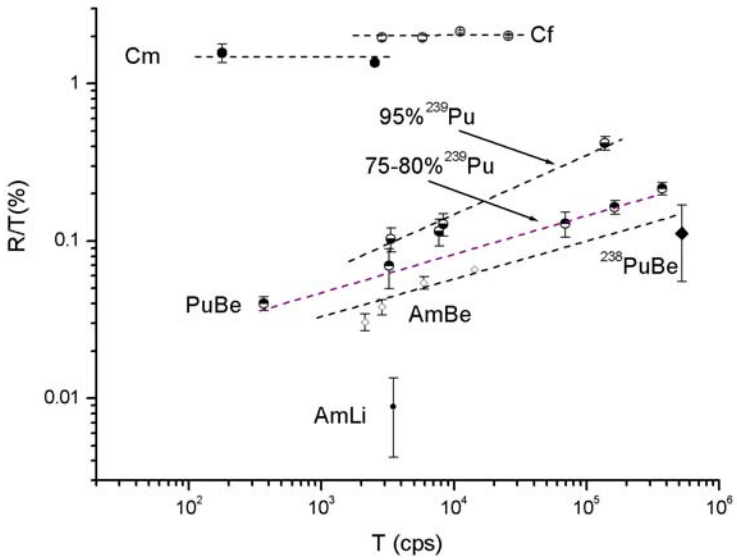

Fig. 5. Ratio *R/T* as a function of total count rate *T* of some transuranium neutron sources measured by the detector of 2.8 % efficiency

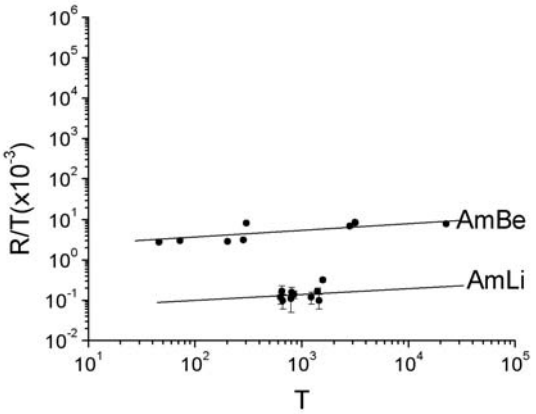

Fig. 6 Ratio *R/T* as a function of *T* - data from Ref. [12]

4. Determination of the transuranium content by gamma-spectrometry

In assaying PuBe neutron sources, gamma-spectrometry plays a very important role. As reported in Refs [1, 2], by measuring the gamma-spectra of PuBe neutron sources in their intense neutron field, the isotopic composition can be determined. Combining the isotopic composition with neutron output, the Pu-content can be evaluated. Pu contents obtained in this way are much smaller than the data still kept in files and reported to safeguards authorities.

In another way, from pure gamma–measurements, correlation of the count rate C_E of a γ-ray of each among the five γ-rays of ^{239}Pu (129, 203, 345, 375, and 413 keV) with the total Pu content, M_{Pu}, can be written as

$$C_E(M_{Pu}) = M_{Pu} f_{239} G_E O_E \frac{1}{F_E} \quad \text{(cps)}, \qquad (12)$$

where f_{239} is the isotopic abundance of ^{239}Pu evaluated from the measured spectrum by the software MGA++, G_E is the specific gamma yield (gamma/g-s), O_E is efficiency of the detector measured by standard reference sources, and F_E is a correction factor for absorption. According to the infinite energy method [3], by plotting $m(1/E)=C_E/(f_{239}*G_E*O_E)$ as a function of $1/E$ for the five γ-rays above, then by extrapolating the curve through five points to $1/E$=0, the value m($1/E$=0) itself gives M_{Pu} of the neutron source.

Comparing the above expression for $m(1/E)$ with formula (12), it is easy to write $m(1/E)=M_{Pu}/F_E$. Considering that Pu-Be sources are all encapsulated in steel cylinders of 3-4 mm thick wall, and they were set far from the detector in order to have a parallel γ-beam arriving at the detector, the absorption correction factor F_E was calculated easily by an approximate model for cylindrical sample, see Fig. 7. In this way, the Pu content was derived from the count rates directly [4], in order to improve the accuracy of the combined method [1, 2].

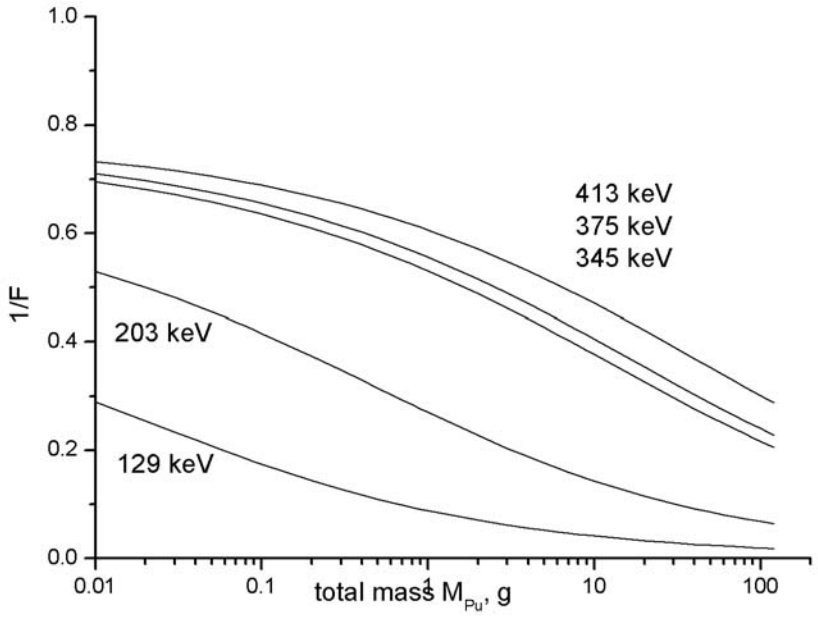

Fig. 7. 1/F as a function of the total Pu content for five γ-rays of ^{239}Pu in a cylindrical configuration

The values for the total Pu-mass in this paper (Fig. 4 and Table 2) were estimated by this method.

5. Conclusions

Using a simple calculation model it was shown that neutron coincidence counting technique can be used for identifying the type and for quantitative assay of transuranium neutron sources, as an independent NDA method. The accuracy and sensitivity of the method mainly depend on the uncertainty of measuring the ratio R/T. Now we are designing an equipment of improved parameters for both low and high intensity neutron sources. The values of the specific neutron yield of each type of neutron sources were also evaluated. The calculations will be extended and refined using Monte Carlo simulations in the near future.

In fact, the ratio *R/T* exhibits not too much variation with the Pu mass. While *R/T* changes an order of magnitude, M_{Pu} spans over three ones. Either *R* or *T* alone would be a more sensitive indicator, thus enabling a more precise calibration. However, depending rather on the neutron output, neither of them varies smoothly or even monotonously with the Pu mass of sources of very different isotopic composition.

The accuracy of the parameters of standard neutron sources plays an important role. In this paper the total Pu contents are estimated on the basis of gamma-spectrometric measurements. Very recently, there was an opportunity to test the accuracy of the method by calorimetry [8]. It turned out that the uncertainty of the pure gamma-spectrometric method is about 10 % for PuBe sources of Pu content below 0.2 g and above 10 g. In the range of 0.5 to 10 g it is 5 to 7 %. An accuracy of about 20 % can be considered a good result for the *R/T* method. For assaying a series of samples it may suffice. This is important in respect of preventing expensive Ge detectors from being damaged in high neutron fields.

The Pu content of PuBe sources in Hungary is going to be determined by the neutron coincidence (R/T) method, calibrated by calorimetry. The whole Pu inventory of these sources will be accounted for on this basis.

Acknowledgements

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References

[1] C. T. Nguyen, J. Bagi, L. Lakosi: *Quantitative NDA of Pu-Be Neutron Sources. Proc 25th Annual Meeting of ESARDA. Symp. on safeguards and nuclear materials management,* Stockholm, 13-15 May 2003, European Commission, Joint Research Centre, EUR 20700 EN, 2003, CD-ROM.

[2] J. Bagi, C. T. Nguyen, L. Lakosi: *Assessment of the Pu content of Pu-Be sources. Nucl. Instr. Meth. B* 222 (2004) 242-248.

[3] J. Morel, M. Vallee, and M. Etcheverry: *Programme de mesure des quantites de plutonium par spectrometrie gamma Logiciel PuMa,* Rapport CEA R-5484, Avril 1989

[4] C. T. Nguyen: *Monitoring Pu-Be neutron sources by gamma-spectrometry. Nucl. Instr. Meth. B,* 2005, submitted

[5] N. Ensslin, J. Stewart, and J. Sapir, *Self-Multiplication Correction Factors for Neutron Coincidence Counting. Nuclear Materials Management* VIII (2), 60 (1979)

[6] M. E. Anderson and W. H. Bond Jr.: *Neutron spectrum of a Plutonium-Beryllium source. Nucl. Phys*. 43 (1963) 330-338

[7] M. E. Anderson and R. A. Neff: *Neutron energy spectra of of different size ^{239}Pu-Be (α,n) sources. Nucl. Instr. Meth*. 99 (1972) 231-235

[8] J. Bagi, L. Lakosi, C. Tam Nguyen, H. Tagziria, B. Pedersen, R. Jaime, P. Schillebeeckx: *Characterisation of PuBe sources by non-destructive assay*, this symposium

9] V. McLane, C. L. Dunford, P. F. Rose: *Neutron cross sections, Vol. 2, Neutron Cross Section Curves,* NNDC, Brookhaven National Laboratory, Upton, N. Y., Academic Press, Boston, 1988
[10] D. M. Cifarelli, W. Hage: *Models for a three-parameter analysis of neutron signal correlation measurements for fissile material assay. Nucl. Instr. Meth. A* 251 (1986) 550-563
[11] L. Lakosi, C. T. Nguyen, J. Bagi, Quantitative NDA *of isotopic neutron sources. Appl. Radiat. Isotopes*, 2005, in press
[12] A. Ravazzani, R, Jaime, M. Looman. B. Pedersen, P. Peerani, P. Schillebeeckx, M. Thornton, A. Foglio Para, and V. Maiorov: *Characterisation of neutron sources by NDA. Proc. 23th Annual Symp. On Safeguards and Nuclear Material Management*, Bruges, Belgium. ESARDA, Rep. EUR 1944 EN, ed. C. Foggi, JRC Ispra, Italy, 2001, pp. 181-191.

Characterization of highly enriched uranium by gamma spectrometry

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Abstract:

A non-destructive gamma-spectrometric procedure is considered to determine the ^{234}U, ^{235}U, ^{238}U, ^{232}U and ^{236}U contents and the age of highly enriched uranium. The ^{234}U and ^{235}U contents were estimated from the 121 and 185.7 keV gamma-rays, respectively. The ^{238}U and ^{232}U contents were evaluated from the gamma-rays of ^{234m}Pa and ^{208}Tl, respectively, but the gamma-spectrum used in this case was measured in a low background chamber. Finally the ^{236}U content was determined from the ratio of the gamma peaks of ^{236}U and ^{235}U. Consequently, the total uranium content and isotopic composition were calculated. The method for age dating was derived on the basis of the daughter/parent relation of $^{214}Bi/^{234}U$, where the ^{214}Bi activity was estimated from the low-background spectrum. The procedure was developed during a "Round Robin" exercise, in which the properties of seized uranium material relevant to nuclear forensics were assessed by several laboratories. The values of the parameters obtained by this procedure are in good agreement with the results obtained using mass-spectrometry by other laboratories, which participated in the "Round Robin" exercise.

Keywords: NDA; low-background gamma spectrometry; nuclear forensics; HEU; isotopic composition

1. Introduction

In 2001 the International Technical Working Group for Combating Illicit Trafficking of Nuclear Material (ITWG) organized an inter-laboratory comparison exercise (Round Robin Exercise - RRE) in order to assess and compare means and methods of forensic analysis on samples of seized nuclear material [1]. In the RRE samples containing about 2 g of a highly enriched uranium oxide powder were sent to the participating laboratories. The participants were asked to determine the properties of this material, relevant for its forensic analysis. The results were expected to be returned to the organisers in periods of 24 hours, 1 week and 2 months.

Among other data, the participants provided a detailed account of the isotopic composition of the material. Some of the laboratories also determined the age of the sample. For precisely determining the isotopic composition and the age of uranium, mass spectrometry is known to be a reliable and accurate method (see e.g. Refs. [2] and [3] for age dating). On the other hand, in the present work it is shown that gamma-spectrometry, when appropriately applied, is suitable for determining this information with an accuracy which is comparable to that of mass spectrometry.

In this paper the gamma-spectrometric methods which have been used in the Institute of Isotopes of the Hungarian Academy of Sciences in order to determine the ^{234}U, ^{235}U, ^{238}U, ^{232}U and ^{236}U contents and the age of the sample received within the Round Robin Exercise are reviewed.

The ^{234}U and ^{235}U content of the sample were determined using a large-area planar high-purity germanium (HPGe) detector. A procedure for self-absorption correction was applied and the count rates per unit mass of the relevant gamma lines of ^{234}U and ^{235}U were determined. The isotope

contents were then calculated by comparing these count rates to the corresponding count rates obtained from the measurement of a reference sample.

A coaxial HPGe detector in a low background iron chamber was used to measure the activities of ^{232}U, ^{238}U and ^{214}Bi, based on the knowledge of the absolute efficiency curve of the detector in the relevant energy interval.

Finally, the activities of ^{236}U and ^{241}Am were estimated from the spectra taken by a medium-area planar HPGe detector, using an intrinsic efficiency calibration method.

From the above measurements the activity ratio ^{214}Bi/^{234}U was calculated, and using this result the age of the sample was also determined. The presence of ^{232}U and ^{236}U indicates that the investigated material was produced in a secondary enrichment process.

2. Determination of the ^{234}U and ^{235}U content

The ^{234}U and ^{235}U contents of the Round Robin Material (RRM) were determined using a 20 cm^2 planar HPGe detector. A reference material (RFM) was used for calibration. The RFM was U_3O_8 powder of known isotopic composition (^{235}U: 90.6 ±1.5 %; ^{238}U: 8.35 ± 0.20 % and ^{234}U: 1.02 ± 0.07%). This means that in 1 g of the RFM there is a total of 0.846 g of uranium consisting of 0.766 ± 0.12 g ^{235}U, 0.070 ± 0.001 g ^{238}U and 0.0086 ± 0.0004 g of ^{234}U. The material was transported to Hungary before 1960, so its age at the time of the measurement (June 2001) was more than 40 years. Both nuclear materials (the RRM and the RFM) were placed separately into a cylindrical polyethylene container of 2.9 cm inner diameter. The spectra of the samples were measured at a 10.8 cm vertical distance above the detector. Four measurements with amounts of approximately 0.5, 1, 1.5 and 2 g were carried out with each material. The count rate of the gamma-transitions at 120.9 keV (^{234}U) and 185.7 (^{235}U) were standardized in unit mass as the ratio, *K(m)*, of the count rate to the total used mass of the sample, *m*, and then they were plotted versus *m* (see Fig. 1).

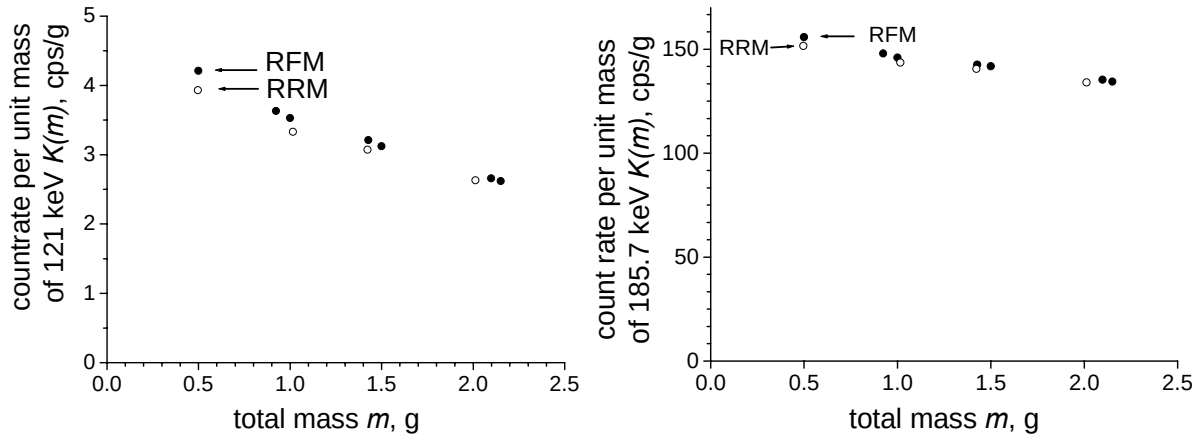

Fig. 1. Count rate of 120.9 and 185.7 keV per unit mass of the sample as a function of the total mass of the sample

The data were fitted with the function

$$K(m) = K_0 \frac{1-e^{-am}}{am}, \tag{1}$$

which models the law of self-absorption within the sample. The parameter K_0 is the value of the curve at *m=0* and physically it represents the net count rate of the 120.9 keV or 185.7 keV gammas of 1 g sample, corrected for self-absorption. The parameter *a* provides information about the matrix of the sample. The ^{234}U and ^{235}U contents of 1 g of the sample were derived by comparing K_0 of the RRM, $K_0(RRM)$, to that of the RFM, $K_0(RFM)$. More precisely, the isotopic contents *M(RRM)* were calculated by inserting the corresponding data into the formula

$$M\,(RRM\,) = M\,(RFM\,)\,\frac{K_o(RRM\,)}{K_o(RFM\,)}\,. \qquad (2)$$

In this way the contents of ^{234}U and ^{235}U in 1 g of the RRM were found to be 0.00807±0005 g/g and 0.754±0.011g/g, respectively.

3. Determining the ^{238}U content and the age of the sample and the activity of ^{232}U using low-background gamma-spectroscopy

3.1. Determination of ^{238}U content

The samples of the RFM (1.05 g) and the RRM (0.975 g) were held in a closed cylindrical polyethylene container of 2.9 cm inner diameter as in the measurements described above and assayed at a distance of 6.5 cm from a 150 cm^3 coaxial HP Germanium detector in a low background chamber. Their gamma-spectrum is shown in Fig. 2a together with the background spectrum.

The gamma-peaks at 766.37 and 1001 keV (Fig. 2b1 and Fig. 2b2) of ^{234m}Pa can be observed clearly. Since ^{234m}Pa is a daughter of ^{238}U, its 1001 keV gamma-transition was used for estimating the ^{238}U content. The count rate per unit mass (K_0) at 1001 keV corresponding to the RRM was compared to the one corresponding to the RFM. Because the sample is “thin” and practically transparent for the 1001 keV line, there was no need to correct for self-absorption. Therefore, the count rate per unit mass (K_0) at 1001 keV was evaluated by simply dividing the measured count rate by the total mass of the sample. Inserting the received K_o values (see Table 1) of the RRM and the RFM to formula (2), the ^{238}U content of 1 g sample was estimated to be 0.065±0.002 g/g.

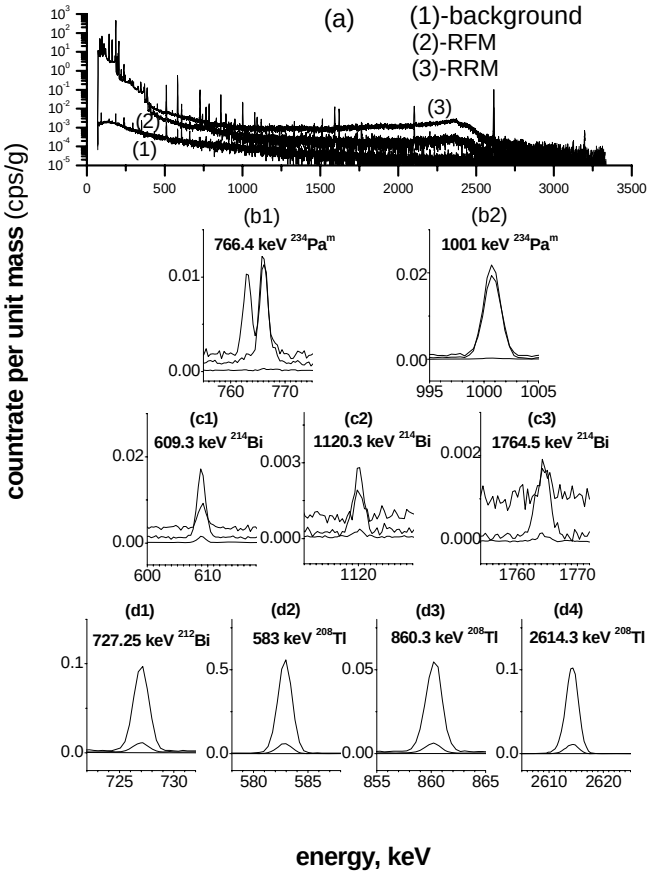

Fig. 2. Gamma-spectrum of the background (1), the reference material (2) and the Round-Robin material (3)

Nucleus	Reference Material			Round-Robin Material		
	Isotope ratio (%)	Content (g/g)	K_0 (cps/g)	K_0 (cps/g)	Content (g/g)	Isotope ratio (%)
^{234}U	1.02 ± 0.04	0.0086 ± 0.0003	4.9 ± 0.15	4.6 ± 0.15	0.00807 ± 0.0004	0.98 ± 0.07
^{235}U	89.5 ± 1.0	0.757 ± 0.0012	163.0 ± 0.7	160.5 ± 0.7	0.745 ± 0.015	89.8 ± 0.7
^{238}U	9.5 ±0.2	0.0804 ± 0.0010	0.00351 ± 0.00003	0.00312 ± 0.00003	0.0715 ± 0.0035	8.6 ± 0.5
^{236}U	m_{236}/m_{235}:	#	#	0.0060 ± 0.0010	0.0051 ±0.0008	0.6 ± 0.1
^{232}U	Activity (Bq/g):	(5.4 ± 0.2)×10^{-11}	44 ± 2	440 ± 10	(5.4 ± 0.1)×10^{-10}	
Total		**0.846**			**0.830 ± 0.020**	

Table 1. Uranium content and isotopic composition of the Reference Material and of the Round-Robin Material

3.2. Age determination

As reported in Ref [4], the activity ratio $^{214}Bi/^{234}U$ can be used as a chronometer for uranium age dating. The gamma-transitions at 609.3, 1120.3 and 1764.5 keV were identified for ^{214}Bi (Fig. 2 c1-c3). The activity of ^{214}Bi in 1g of the sample is calculated by the formula:

$$A = \frac{K}{B\varepsilon}, \tag{3}$$

where K is the count rate of 1g of the sample (cps/g), B is the decay branching [5] and ε is the detector efficiency measured by point-like standard sources. Because of using a thin sample, the count rate does not need to be corrected for self-absorption in the sample.

The activity of ^{214}Bi was calculated in Ref. [4] and it is summarized in Table 2. ^{214}Bi is a daughter of ^{234}U, which decays through ^{230}Th to ^{226}Ra, which, in turn, decays to ^{214}Bi through three short-lived nuclides. If the activity of ^{214}Bi, A, and the activity of ^{234}U, A_0, are known, the age T of the sample can be derived from the following formula [4]:

$$\frac{A}{A_0} = \frac{1}{2}\lambda_2\lambda_3 T^2, \tag{4}$$

where λ_2 =0.288x10^{-12} s^{-1} and λ_3 = 13.86x10^{-12} s^{-1} are the decay constants of ^{230}Th and ^{226}Ra [5], respectively. The age of the RRM was estimated in this way to be 23±3 years. The age of the RFM was also calculated and it was found to be 42±3 years, which was the expected value.

3.3. Activity of ^{232}U

The gamma-peaks at 583, 860.3 and 2614.3 keV (Fig. 2 d2-d4) were identified for ^{208}Tl. Using the parameters listed in Table 2, ^{208}Tl activity was estimated in the same way as the activity of ^{214}Bi and it was found to be 158±5 and 15.1±0.5 Bq/g for the RRM and the RFM, respectively (see Table 2), as the weighted average of the values obtained from each transition.

The agreement of ^{208}Tl activities calculated from the above gamma-lines proves that their origin is indeed ^{208}Tl present in the measured sample. The value of the ^{208}Tl activity was confirmed by the measurement of the activity of the ^{208}Tl-parent ^{212}Bi using the gamma-line at 727.25 keV (Fig. 2 d1). ^{212}Bi decays to ^{208}Tl with branching ratio of 0.36 [5]. Its activity was 438±10 and 43±2 Bq/g for the RRM and the RFM, respectively (see Table 2), which were the expected values.

In the investigated material ^{212}Bi and ^{208}Tl may have their origin either from the decay of ^{232}Th or ^{232}U [5], but the absence of the gamma-rays at 911.16 and 968.97 keV of ^{228}Ac (a daughter of ^{232}Th) in the

measured spectrum suggests that their origin is the decay of ^{232}U. The measured ages of the samples were more than 10 times longer than the half-life of ^{208}Th, which has the longest half-life in the ^{232}U decay chain, so the samples were in equilibrium. Therefore, ^{232}U activities were simply calculated from the activities of the ^{232}U daughters (^{212}Bi and ^{208}Tl) and found to be 440±10 and 44±2 Bq/g corresponding the ^{232}U content of (5.4±0.2)x10^{-10} and (5.4±0.1)x10^{-11} g/g for the RRM and the RFM, respectively.

Nucleus	Energy (keV)	Decay branching [5] (%)	Count rate per unit mass (×100, cps/g)	Detector efficiency (%)	Activity per unit mass (Bq/g)
^{234}U RRM	120.9	0.0342	460 ± 15	0.730 ± 0.015	**(1.84 ±0.06)×10^6 [4]**
RFM			490 ± 15		**(1.91 ± 0.06) ×10^6 [4]**
^{214}Bi RRM	609.3,			**Average:**	**2.0 ± 0.25 [4]**
RFM	1220.3, 1764			**Average:**	**7.06 ± 0.15 [4]**
^{212}Bi RRM	727.3	6.65	14.1 ± 0.05	0.483 ± 0.01	438 ± 10
RFM			1.38 ± 0.05		43 ± 2
^{208}Tl RRM	583.0	86	76.1 ± 0.2	0.58 ± 0.01	155 ± 3
	860.3	12.0	8.38 ± 0.09	0.45 ± 0.01	159 ± 5
	2614.3	99.79	25.4 ± 0.05	0.15 ± 0.02	169 ± 17
				Average:	**158 ± 5**
RFM	583.0	86	7.72 ± 0.15	0.58 ± 0.01	15.5 ± 0.3
	860.3	12.0	0.76 ± 0.05	0.45 ± 0.01	14.3 ± 0.7
	2614.3	99.79	2.46 ± 0.05	0.15 ± 0.02	16.4 ± 1.7
				Average:	**15.1 ± 0.5**

Table 2. Activity of ^{234}U, ^{214}Bi, ^{212}Bi and ^{208}Tl

4. Determination of ^{236}U content and ^{241}Am activity by a medium area planar HPGe detector

The 49.37 keV peak observed in the spectrum was identified for ^{236}U decay. The ratio of ^{236}U and ^{235}U contents was evaluated by comparing the yield of this peak to the yield curve obtained from the yields of the 58.6, 84.2 and 90 keV peaks of ^{235}U and it was found to be (0.67±0.10)x10^{-2}. Then the ^{236}U content was calculated to be (0.51±0.08)x10^{-2} g/g for the RRM.

Similarly, the 59.6 keV line was identified for ^{241}Am and its activity was calculated by peak ratio technique as above and found to be 125±15 Bq/g for the RRM.

5. Results and discussion

The procedures described in this work were developed during the Round-Robin Exercise with highly enriched uranium [1]. In the first report to the organizers of the exercise (24 hours), the procedures described here were used for estimating the ^{235}U content of the material and the total uranium content. In the 24-hour report the isotopic composition determined by the MGA++ commercial software [6] was included, based on the analysis of the gamma spectra taken by a medium area planar HP-Germanium detector. In the second report (1 week), the values of isotopic composition obtained by two different methods were sent: one from analyzing the material by mass-spectrometry and the other by MGA++. In the final report (2 months), the age of the Round-Robin material and its ^{232}U and ^{236}U content determined by the methods described here were also included.

In the present paper independent NDA methods for HEU characterization were described by characterizing the nuclear material from the Round Robin Exercise. The results are compared to the results obtained by mass spectrometry in Table 3.

Age dating is probably the most interesting result of the presented research. In the Round Robin Exercise the age of HEU was estimated by gamma spectrometry for the first time. This method does

not require the use of calibration standards and its uncertainty is comparable to that of mass-spectrometry (see Table 3).

Age dating requires measuring the content of ^{234}U and ^{214}Bi as precisely as possible. Unlike commercially available software ([6], [7], [8]), the procedure described in this work can provide a quantitative estimation of the ^{234}U content which is sufficiently precise for age dating.

Laboratory nickname	U-234	U-235	U-236	U-238	U-232(g/g)	Age(years)	
Azores	0.97	89.99	0.68	8.37		22,2-22,6	MS
Barbados		85.6 ± 3.8					GS
Borneo	0.85 ± 0.15	86.7 ± 1.5	0.57 ± 1.08	11.9 ± 0.9			MS
Chatham	0.960 ± 0.001	89.94 ± 0.06	0.0643 ± 0.003	8.462 ± 0.006			MS
Galapagos	0.96	89.89	0.68	8.47			MS
Mindanao	0.96 ± 0.40	89.9 ± 0.11	0.678 ± 0.23	8.443 ± 1.29		22,4 ± 1,2	MS
Tobago	1.05 ± 0.07	89.37 ± 1.8	0.69 ± 0.05	8.88 ± 0.2			MS
Tonga	0.967 ± 0.001	89.99 ± 0.02	0.679 ± 0.001	8.362 ± 0.005			MS
Trinidad	0.955 ± 0.075	90.01 ± 0.35	0.673 ± 0.030	8.365 ± 0.033		23.5 ± 0.5	MS
Present work	**0.98 ± 0.07**	**89.8 ± 0.7**	**0.6 ± 0.1**	**8.6 ± 0.5**	**(5.4 ± 0.1)×10^{-10}**	**23 ± 3**	**GS**

Table 3. Isotopic composition, ^{232}U content and the age of the Round-Robin Material, measured by different laboratories using mass-spectrometry (MS) [1] and gamma-spectrometry (GS)

The estimation of total uranium content is an important result of this analysis. The obtained value of 0.835 ± 0.015 g/g is close to the uranium content of U_3O_8 (0.846 g/g). But if we note that the amounts of the three uranium isotopes (^{234}U, ^{235}U and ^{238}U) in the RRM are significantly smaller than in the RFM (Table 1 or Fig. 1. a,b and Fig. 2 b2), which is known to be U_3O_8, it follows that the matrix of the RRM is different from the matrix of the RFM and it cannot be identified with U_3O_8.

The ability to determine a small ^{238}U amount in a low background chamber makes it possible to precisely determine the isotopic composition of a material even with nearly 100% of ^{235}U.

Estimating the ^{232}U and ^{236}U content with high sensitivity and accuracy provides important information about unknown nuclear material. Because the nuclides of ^{232}U and ^{236}U are produced during the operation of a nuclear reactor, their existence in the investigated sample shows that the HEU material was produced in a secondary separation process. A good energy resolution planar HP-Germanium (small or medium area) should be used to observe the 49.37 keV peak of ^{236}U for improving the accuracy of the result.

In conclusion, the methods presented here are powerful tools of NDA for the characterization of highly enriched uranium.

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References

[1] Dudder, G. B., Hanlen, R. C., Herbillon, G. H., *HEU RR Status*, Presentation given at the Seventh Meeting of the International Technical Working Group for Combating Illicit Trafficking of Nuclear Materials, June 27-28, 2002, Luxembourg

[2] Wallenius, M., Morgenstern, A., Apostolidis, C., Mayer, K., *Determination of the age of highly enriched uranium*, *Anal. Bioanal. Chem*. **374**, (2002), 379-384

[3] Morgenstern, A., Apostolidis, C., Mayer, K., *Age Determination of Highly Enriched Uranium: Separation and Analysis of ^{231}Pa*, *Anal. Chem*. **72** (2002) 5513-5516
[4] Tam Nguyen C., *Age-dating of highly enriched Uranium by gamma-spectrometry*, *Nucl. Instrum. Meth. B* **229** (2005) 103-110
[5] Browne, E., Firestone, R. B, *Table of Radioactive Isotopes*, Wiley, New York (1986)
[6] MGA++ Software User's Manual, Software Version 1, EG&G ORTEC
[7] Clark, D., *U235: A Gamma Ray Analysis Code for Uranium Isotopic Determination*, UCRL-ID-125727, (1996)
[8] Gunnink R. *et al*, *MGAU: A New Analysis Code for Measuring U-235 Enrichments in Arbitrary Samples*, UCRL-JC-114713, (1994)

Generalized Approach of the IDGS Technique

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Abstract

Isotope Dilution Gamma-ray Spectrometry (IDGS) was originally developed to measure plutonium isotopic compositions and concentrations in spent-fuel dissovler solutions at a reprocessing plant. The isotopic compositions are determined by high-resolution gamma-ray spectroscopy. To measure the plutonium concentration, the unknown solution is spiked with plutonium of accurately known plutonium mass and isotopic composition. And the concentration of plutonium in the unknown solution is then traditionally determined by calculating the differences among the ^{239}Pu weight percent (wt %) and isotopic ^{240}Pu/^{239}Pu ratios of the spike, the spiked solution, and the unknown (un-spiked) solution.

To improve the IDGS capability, we are developing a generalized approach of IDGS technique that allows the calculation of the concentration in the unknown solution using other plutonium isotopic weight percents and isotopic ratios in addition to the ^{239}Pu wt % and ^{240}Pu/^{239}Pu ratio.

*The generalized equation, which is (**i, j**) symmetrical, for the concentration of plutonium in the unknown solution then can be written as*

$$\mathrm{C(Pu)}^{ij} = \frac{M_s}{V_u} \bullet \frac{W_s^i}{W_u^i} \bullet \frac{R_m^{ij} - R_s^{ij}}{R_u^{ij} - R_m^{ij}},$$

Where

- M_s = *mass of plutonium in the spike*
- V_u = *volume of dissolver solution taken*
- W_s^i = *weight fraction of isotope **i** in the spike*
- W_u^i = *weight fraction of isotope **i** in the unknown solution*
- R_m^{ij} = *the ratio of isotope **j** to isotope **i** in the spiked solution*
- R_s^{ij} = *the ratio of isotope **j** to isotope **i** in the spike*
- R_u^{ij} = *the ratio of isotope **j** to isotope **i** in the unknown solution*
- ***i**, **j*** = *plutonium or americium isotopes (^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{241}Am)*

The results obtained by this generalized IDGS technique are about a factor of two better than the method employing the ^{240}Pu/^{239}Pu ratio alone for both spent-fuel dissolver solution and product solutions.

Keywords: Isotope Dilution Gamma-ray Spectrometry (IDGS), Gamma-ray, Spectrometry, plutonium, isotopic.

1. INTRODUCTION

Isotope Dilution Gamma-ray Spectrometry (IDGS) was originally developed to measure simultaneously plutonium isotopic compositions and concentrations in spent-fuel dissovler solutions at a reprocessing plant. The isotopic compositions are determined by high-resolution gamma-ray spectroscopy. To measure the plutonium concentration, the unknown solution is spiked with plutonium of accurately known plutonium mass and isotopic composition. And the concentration of plutonium in the unknown solution is then traditionally determined by calculating the differences among the ^{239}Pu weight percent (wt %) and isotopic ^{240}Pu/^{239}Pu ratios of the spike, the spiked solution, and the unknown (un-spiked) solution. This is because ^{239}Pu and ^{240}Pu are major isotopes in both spent-fuel dissolver solutions and spikes.

$$\mathrm{C(Pu)} = \frac{M_s}{V_u} \bullet \frac{W_s^9}{W_u^9} \bullet \frac{R_m - R_s}{R_u - R_m}, \qquad (1)$$

where

M_s = mass of plutonium in the spike
V_u = volume of dissolver solution taken
W_s^9 = weight fraction of ^{239}Pu in the spike
W_u^9 = weight fraction of ^{239}Pu in the unknown solution
R_m = the ^{240}Pu/^{239}Pu ratio in the mixed or spiked solution
R_s = the ^{240}Pu/^{239}Pu ratio in the spike
R_u = the ^{240}Pu/^{239}Pu ratio in the unknown solution

In this equation, the values of M_s, V_u, W_s^9, and R_s are known. Only the values of R_u and W_u^9 in the unknown solution and R_m in the spiked solution are needed to be measured by gamma-ray spectrometry [1,2].

Equation 1 utilizes only the isotopic ^{240}Pu/^{239}Pu ratios of the spike, the spiked solution, and the unknown solution. To enhance the IDGS capability, we are developing a generalization of the IDGS technique that the calculation of the concentration in the unknown solution using other plutonium isotopic weight percents and isotopic ratios in addition to the ^{239}Pu wt % and ^{240}Pu/^{239}Pu ratio.

This paper will discuss the result of analysis on input spent-fuel dissolver solutions by using generalized IDGS technique.

2. THE GENERALIZED APPROACH

The generalized equation for the concentration of plutonium in the unknown solution then can be written as

$$\mathrm{C(Pu)}^{ij} = \frac{M_s}{V_u} \bullet \frac{W_s^i}{W_u^i} \bullet \frac{R_m^{ij} - R_s^{ij}}{R_u^{ij} - R_m^{ij}}, \qquad (2)$$

Where

M_s = mass of plutonium in the spike
V_u = volume of dissolver solution taken
W_s^i = weight fraction of isotope ***i*** in the spike
W_u^i = weight fraction of isotope ***i*** in the unknown solution
R_m^{ij} = the ratio of isotope ***j*** to isotope ***i*** in the spiked solution
R_s^{ij} = the ratio of isotope ***j*** to isotope ***i*** in the spike
R_u^{ij} = the ratio of isotope ***j*** to isotope ***i*** in the unknown solution
i, ***j*** = plutonium or americium isotopes (^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{241}Am)

Note that when i = ^{239}Pu and j = ^{240}Pu, Eq. (2) is exactly same as Eq. (1). And the equation (2) is (***i, j)*** symmetrical. That is,

$$C(Pu)^{ij} = \frac{M_s}{V_u} \bullet \frac{W_s^i}{W_u^i} \bullet \frac{(R_m^{ij} - R_s^{ij})}{(R_u^{ij} - R_m^{ij})} = \frac{M_s}{V_u} \bullet \frac{W_s^i}{W_u^i} \bullet \frac{\left(1/R_m^{ji} - 1/R_s^{ji} \right)}{\left(1/R_u^{ji} - 1/R_m^{ji} \right)} = \frac{M_s}{V_u} \bullet \frac{W_s^i}{W_u^i} \bullet \frac{\left((R_s^{ji} - R_m^{ji}) / R_m^{ji} R_s^{ji} \right)}{\left((R_m^{ji} - R_u^{ji}) / R_u^{ji} R_m^{ji} \right)}$$

$$= \frac{M_s}{V_u} \bullet \frac{W_s^i}{W_u^i} \bullet \frac{R_u^{ji}}{R_s^{ji}} \bullet \frac{(R_m^{ji} - R_s^{ji})}{(R_u^{ji} - R_m^{ji})} = \frac{M_s}{V_u} \bullet \frac{W_s^i}{R_s^{ji}} \bullet \frac{R_u^{ji}}{W_u^i} \bullet \frac{(R_m^{ji} - R_s^{ji})}{(R_u^{ji} - R_m^{ji})} = \frac{M_s}{V_u} \bullet \frac{W_s^j}{W_u^j} \bullet \frac{(R_m^{ji} - R_s^{ji})}{(R_u^{ji} - R_m^{ji})} = C(Pu)^{ji}$$

In some processes, the mass of the unknown solution is used instead of the volume. Then the plutonium concentration C_M (Pu) will have the unit of grams (or mg) of plutonium per unit mass of the unknown solution instead of grams of plutonium per unit volume of the unknown solution.

Recently, we have tested this generalized IDGS Equation 2 by analyzing the production fuel plutonium gamma-ray data taken at the Plutonium Fuel Center, Japan Nuclear Cycle Development Institute (JNC), Tokai, Japan [3]. We found, from that study, that even though we can use the ratios of all the isotopes to determine the concentration of plutonium in the solution, we do not need to. We only need the ratios of all the isotopes to that of ^{239}Pu in the calculations and the result will be almost the same as if all the isotopes have been used. That is, the index ***i*** in Equation 2 is fixed to the ^{239}Pu isotope.

We analyzed two different types of data, aged and fresh plutonium, and found that the results from this generalized IDGS technique are almost a factor of three and two better than those of the traditional IDGS for the aged and fresh plutonium, respectively [3]. We are now applying that same generalized IDGS technique to the analysis of spent-fuel dissolver solutions.

3. SPENT-FUEL DISSOLVER SOLUTIONS

Instead of obtaining new spent-fuel data for this generalized IDGS study, we are using raw spectra from published data to show that the technique can significantly improve the spent-fuel results. The data were obtained in 2001 from the study of the different spent-fuel sample separation schemes to improve the IDGS analysis done at the Tokai Reprocessing Center (TRC), Japan Nuclear Cycle Development Institute (JNC) [4].

3.1. Sample preparation

In that study, the extraction chromatography using U/TEVA•Spec resin [5] (for uranium and tetravalent actinides specifically) was used to purify and recover both plutonium and uranium from dissolver solutions. The separation scheme for preparing the sample by using the extraction chromatography is shown in Figure 1.

About 1 ml aliquot each for the spiked samples and the unspiked samples were taken from dissolver solutions, and weighted by a precise electronic balance. The spiked samples were prepared by dissolving the dissolver solution (1 ml aliquot) with large size dry (LSD) spikes at 90°C on a heater and then mixing them by a magnetic stirrer.

Plutonium in these samples was completely adjusted to tetra-valence with Fe(II) and $NaNO_2$, and they were dissolved again with 8M-HNO_3: (1 ml) after heating them to near-dryness at 90°C. Each sample was individually passed through the extraction chromatographic columns (U/TEVA • Spec resin) adjusted with

8M-HNO_3, fission products and americium were separated from uranium and plutonium by washing with 8M-HNO_3, 3M-HNO_3, and 0.01M-HNO_3. Uranium and plutonium were eluted with 0.01M-HNO_3. Eluted

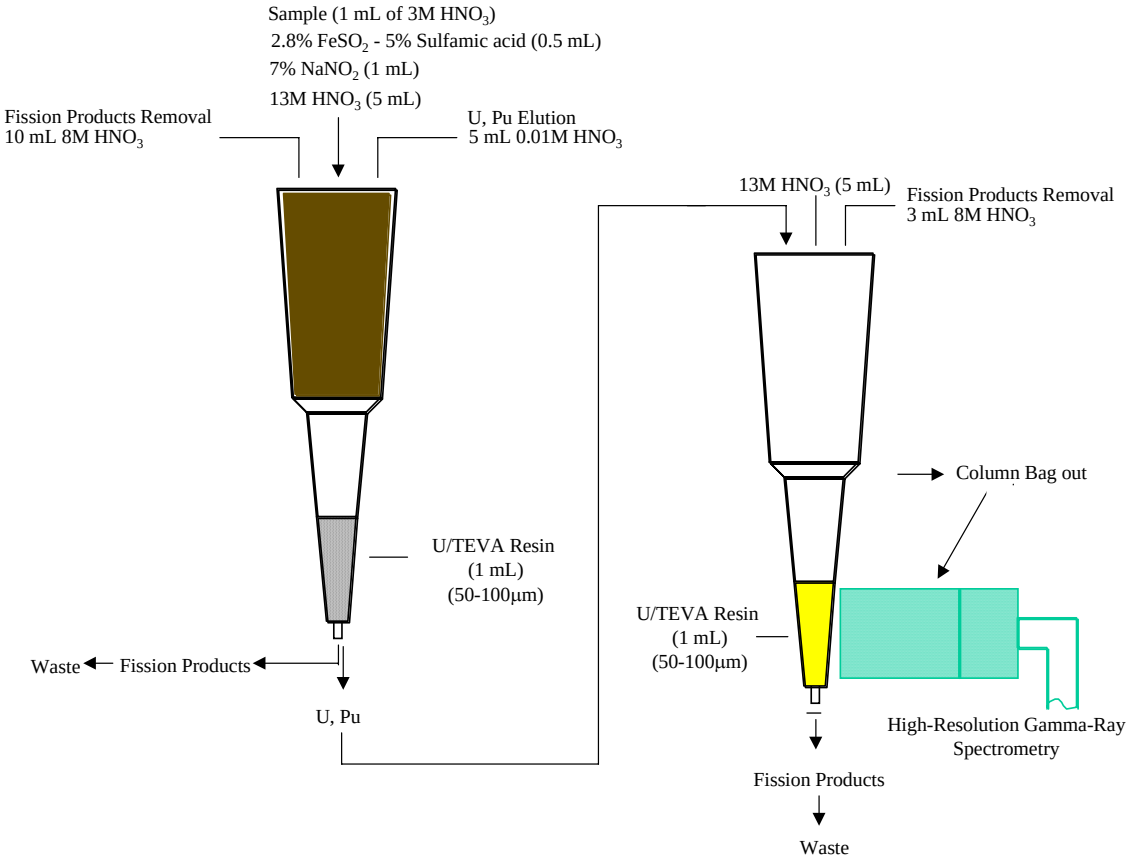

Figure 1. Plutonium and Uranium separation scheme using U/TEVA•Spec® resin

ones were absorbed in the columns again, and fission products like ruthenium were separated by washing with 3M-HNO_3. Because gamma rays of fission products have possibly influencing the gamma-ray measurements if significant amount of fission products were still contained in the samples. Each sample was carefully removed not to contaminate with a plastic bag from a glove box and measured by a high-resolution gamma-ray spectrometry (HRGS).

We refer the separation scheme discussed above as Scheme O. To improve and simplify the sample preparation, we have considered and examined several optional separation schemes as follows:

Scheme A: Same as Scheme O discussed above and shown in Fig. 1, except heat at 90°C for 30 min. after valency adjustment with $NaNO_2$.

Scheme B: Same as Scheme O discussed above and shown in Fig. 1, except heat at 90°C for 10 min. after valency adjustment with $NaNO_2$.

Scheme C: First part of Scheme A. No elution with 0.01M-HNO_3 and the second fission products re-washing in Scheme A. Bag out the column for HRGS measurement after the first fission products removal. Scheme C is shown in Fig. 2.

Scheme D: First part of Scheme B. No elution with 0.01M-HNO_3 and the second fission products re-washing in Scheme B. Bag out the column for HRGS measurement after the first fission product's removal. Scheme D is shown in Fig. 2.

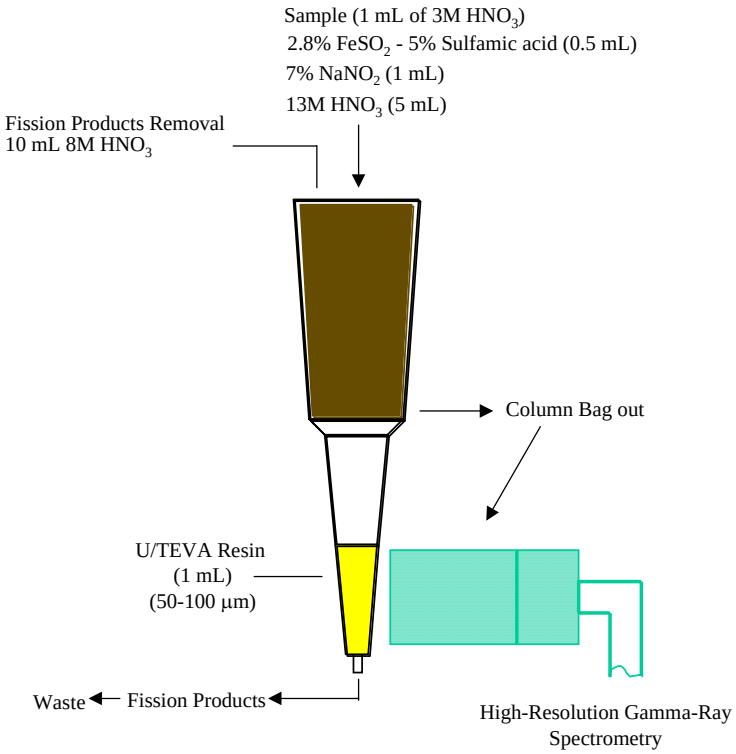

Figure 2. Simplified separation scheme using U/TEV•Spec® resin.

3.2. Analysis

The generalized IDGS technique employs the ratios of many isotopes including americium and uranium to that of ^{239}Pu for the determination of the plutonium concentration [3]. However, for freshly separated samples such as the spent-fuel dissolver solutions, because plutonium, uranium, and americium concentrations are altered differently during the separation process, we will not be able to use uranium or americium, but only plutonium, to determine the plutonium concentration. That means we can only use the isotopic ratios of $^{238}Pu/^{239}Pu$, $^{240}Pu/^{239}Pu$, and $^{241}Pu/^{239}Pu$ in the calculations of the plutonium concentration. A typical low-energy high-resolution gamma-ray spectrum of spent-fuel dissolve solution after chemical separation with extraction chromatography using U/TEVA•Spec resins is shown in Fig. 3.

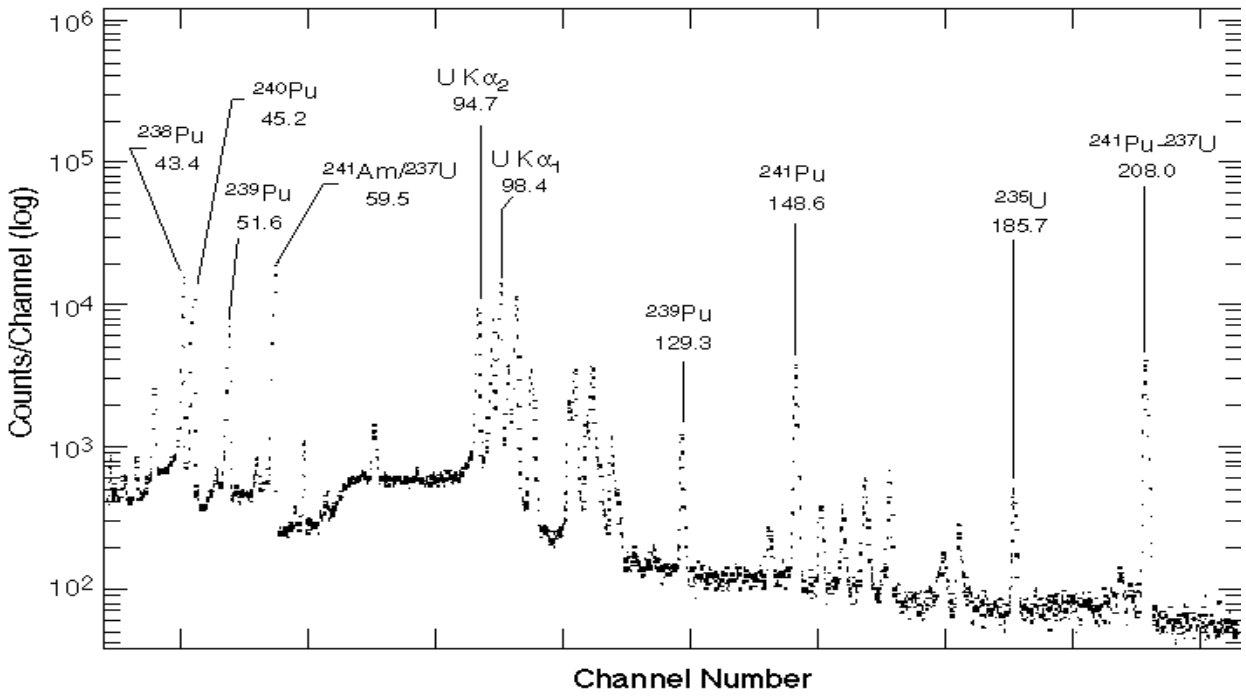

Figure 3. Gamma-ray spectrum of spent-fuel dissolve solution after chemical separation with extraction chromatography using U/TEVA•Spec resins.

Two different batches of dissolver solutions were made and measured by both gamma ray spectroscopy and mass spectrometry done at the Tokai Reprocessing Center, Japan Nuclear Cycle Development Institute [4]. Table 1 shows the results obtained by mass spectrometry.

Table 1. Isotopic compositions of the solutions as determined by mass spectrometry

Solution	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu
1	1.307	58.954	27.554	7.105	5.080
2	1.468	62.042	23.707	8.095	4.688

Table 2 shows the ratios of the gamma ray spectroscopy (GS) results to that of the mass spectrometry MS). The FRAM code (Fixed-Energy Response-Function Analysis with Multiple Efficiency) version 4 [6,7] with the parameter set ***Upu38_210SpFuel*** was used to analyze the data. The errors are of the gamma ray measurements alone (i.e. the errors from the mass spectrometry are assumed to be insignificant comparing with that of the gamma ray measurements). The samples #3 (scheme B) and #5 (scheme C) were taken from the dissolver solution 1 (with the mass spectrometry isotopic compositions shown in the Table 1) and the other samples were taken from the dissolver solution 2.

Table 2. Comparison of plutonium isotopic composition as determined by gamma isotopic analysis (GS) and mass spectrometry (MS).

Sample	Scheme	^{238}Pu		^{239}Pu		^{240}Pu		^{241}Pu		^{242}Pu	
		GS/MS	%err	GS/MS	%err	GS/MS	%err	GS/MS	%err	GS/MS	%err
1	O	1.0010	0.32	0.9962	0.13	1.0030	0.29	1.0103	0.33	1.0176	0.50
2	A	0.9899	0.52	1.0027	0.21	0.9952	0.47	0.9991	0.49	0.9935	0.78
3	B	0.9958	0.82	0.9965	0.38	0.9991	0.72	1.0114	1.03	1.0304	1.29
4	B	0.9982	0.41	0.9990	0.17	0.9999	0.38	1.0041	0.41	1.0079	0.64
5	C	0.9923	0.57	1.0028	0.25	0.9898	0.48	1.0096	0.55	1.0114	0.87
6	D	0.9943	0.57	1.0008	0.23	0.9942	0.51	1.0113	0.55	1.0008	0.88

Plutonium-242 does not have any measurable gamma ray so it cannot be determined directly from gamma ray measurements. Its value, however, can be determined with reasonable accuracy from its correlations with other isotopes in the sample. The ^{242}Pu gamma ray results used in the Table 2 were obtained by correlation. Their errors were from statistical propagations only and do not include the correlation errors.

In normal course of studies of the gamma ray isotopic analysis of plutonium, one normally omits the ^{242}Pu comparison or set it equal to the accepted value (normally determined by mass spectrometry). However, in a real IDGS measurement where it completely replaces MS then there would be no mass spectrometry value to fix to. Then the ^{242}Pu value will have to be determined by correlations and accepted into the IDGS calculations. That is the reason why we include the correlation results of ^{242}Pu in this study.

Table 3 shows the plutonium element concentrations determined from different components of the generalized IDGS and of Isotope Dilution Mass Spectrometry (IDMS).

Table 3. Comparison of plutonium isotopic composition as determined by gamma isotopic analysis and mass spectrometry.

Sample	Scheme	**IDGS** (g Pu/l)								**IDMS**	**IDGS/IDMS**			
		^{238}Pu	%err	^{240}Pu	%err	^{241}Pu	%err	All 3	%err		^{238}Pu	^{240}Pu	^{241}Pu	All 3
1	O	1.564	0.80	1.586	0.99	1.578	0.89	1.574	0.51	**1.584**	0.987	1.001	0.996	0.994
2	A	1.597	1.12	1.592	1.34	1.608	1.17	1.599	0.69	**1.584**	1.008	1.005	1.015	1.010
3	B	1.278	1.59	1.286	1.85	1.268	1.92	1.278	1.02	**1.276**	1.001	1.008	0.993	1.001
4	B	1.581	0.89	1.584	1.06	1.593	0.95	1.586	0.55	**1.584**	0.998	1.000	1.006	1.001
5	C	1.284	1.18	1.316	1.36	1.264	1.25	1.286	0.73	**1.276**	1.007	1.031	0.990	1.008
6	D	1.588	1.13	1.599	1.31	1.569	1.16	1.584	0.69	**1.584**	1.002	1.010	0.990	1.000

Each pair of columns from column 3-8 shows the individual IDGS results for ***j*** equals to ^{238}Pu, ^{240}Pu, and ^{241}Pu (with ***i*** set to ^{239}Pu) in Equation 2. The next pair of columns shows the combined results from all three isotopes.

We see that the reported errors (which comes from statistics only) agrees reasonably well with the measured bias. Also, the results from the combination of all three isotopes appear to be have errors about half of those from the calculations of ^{240}Pu alone, which is the traditional IDGS.

4. DISCUSSION

We just have shown that the generalized IDGS technique work well with various spent-fuel separation schemes done at the Tokai Reprocessing Center, Japan Nuclear Cycle Development Institute (JNC). The ^{242}Pu correlation appears to give reasonable results for the types of material used there too. This would help with reducing the bias from IDGS calculations.

Note that since ^{242}Pu cannot be determined directly from gamma-ray measurements, we do not use it in the calculations of the plutonium concentration. However, even though it is not used directly in the calculations, it can affect the plutonium concentration result and its error indirectly. Equation 2 has the weight fraction of isotope ***i*** (or ^{239}Pu) in the unknown solution term W_u^i. A 10% error for ^{242}Pu in the unknown solution, which may not be unreasonable since ^{242}Pu is determined from correlation and the correlation may not work well with the type of measured material, can translate into about 0.5% errors in ^{239}Pu and any other isotopes.

5. ACKNOWLEDGMENT

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6. REFERENCES

[1] T. K. Li, Y. Kuno, S. Sato, K. Nakatsuka, and T. Akiyama, “Determination of Plutonium Concentration and Isotopic Compositions by Isotope Dilution Gamma-Ray Spectroscopy on Resin Beads,” *Nucl. Mater. Manage*. XIX (Proc. Issue), 390-396 (1990).

[2] T. K. Li, J. L. Parker, Y. Kuno, S. Sato, A. Kurosawa, and T. Aliyama, “Development of Isotope Dilution Gamma-Ray Spectrometry for Plutonium Analysis,” *Proc. 13th ESARDA Symposium on Safeguards and Nucl. Mater. Manage.* (ESARDA, Avignon, France, 1991) 24, pp. 175-185.

[3] Duc T. Vo and Tien K. Li, “Generalization of the IDGS technique,” Los Alamos National Laboratory report LA-UR-04-4186, Proceeding of the INMM 45th Annual Meeting, Orlando, Florida, USA, 2004.

[4] T. K. Li, M. Watanabe, S. Sato, S. Jitsukata, T. Kuno, H. Ikeda, and T. Hiyama, “Improvement in Measurements of Plutonium in Spent-Fuel Dissolver Solutions,” Los Alamos National Laboratory report LA-UR-01-3537, Proceeding of the INMM 42nd Annual Meeting, Indian Wells, California, USA, 2001.

[5] T. K. Li, T. Kuno, O. Kitagawa, S. Sato, A. Kurosawa, and Y. Kuno, “Simultaneous Measurements of Plutonium and Uranium in Spent-Fuel Dissolver Solutions,” Los Alamos National Laboratory report LA-UR-97-2715.

[6] T. E. Sampson, G. W. Nelson, and T. A. Kelley, “FRAM: A Versatile Code for Analyzing the Isotopic Composition of Plutonium from Gamma-Ray Pulse Height Spectra,” Los Alamos National Laboratory report LA-11720-MS (1989).

[7] T. E. Sampson, T. A. Kelley, and D.T. Vo, “Application Guide to Gamma-Ray Isotopic Analysis Using the FRAM Software,” Los Alamos National Laboratory report LA-14018 (2003).

NUSIMEP 4: Measurement of Uranium Isotopic Abundances in a Simulated Urine Solution

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Abstract:

The previous NUSIMEP ('Nuclear Signatures Inter-laboratory Measurement Evaluation Programme') campaigns were designed for the measurement of uranium isotopic ratios: in a simple nitrate matrix (NUSIMEP 2) and in a saline solution (NUSIMEP 3). For NUSIMEP 4 it was decided to prepare the uranium in a more realistic matrix, a simulated urine solution.

As for the previous campaigns, the uranium samples were prepared by mixing UF_6 in the gas phase and measuring $n(^{235}U)/n(^{238}U)$ in a gas-source mass-spectrometer. Following this the minor isotopic ratios were measured by TIMS calibrated internally by the certified ratio $n(^{235}U)/n(^{238}U)$.

The matrix solution was prepared from a mixture of salts to produce a 2 % saline solution. This was chemically purified from uranium and the remaining trace quantity of this element was measured by ICP-MS. The (natural) uranium in a sample of urea was also measured. The concentrated uranium solutions prepared by hydrolysing the UF_6 samples were diluted in cleaned quartz flasks with 1 M HNO_3 prepared from sub-boiled nitric acid and mixed with the saline solution and urea to make two solutions with approximately 5 ng $U \cdot g^{-1}$ solution. The uranium isotopic vector was certified from the measurements as UF_6 and by TIMS, and taking into account the small contribution from blanks from matrix and equipment.

The solutions were transported to the participants' laboratories in cleaned polypropylene bottles. Each laboratory was requested to measure the ratios $n(^{234}U)/n(^{238}U)$, $n(^{235}U)/n(^{238}U)$, $n(^{236}U)/n(^{238}U)$ and to report the measured values with an uncertainty preferably following the ISO/GUM guides.

The collected results show a picture of the present status of measurements of uranium isotopes on this level of uranium. A parallel campaign conducted by the CCQM, P-48, will collect measurement results from selected laboratories for similar materials and these results will subsequently be compared with the NUSIMEP 4 results.

Keywords: *Uranium isotopic content, simulated biological matrix, Inter-laboratory comparison*

1. Introduction

Uranium is an analyte of high importance in the fields of safeguards, fissile material accountancy and control and also in nuclear non-proliferation. The NUSIMEP 4 campaign was conceived as a continuation of the previous campaigns devoted to measurements of uranium isotopic ratios in various matrices. NUSIMEP 2 was designed to test the measurement of uranium isotope ratios in a simple (dried nitrate) matrix and for NUSIMEP 3 a saline solution as a more challenging matrix was employed. In the present campaign a simulated urine was chosen as the matrix. The analysis of the uranium isotopic content in this medium is important for detecting and measuring exposure to uranium, for instance to measure the exposure to depleted uranium in war zones where munitions with depleted uranium metal were used.

Again, as for NUSIMEP 2 and NUSIMEP 3, the intention is to prepare samples for testing the measurement of the uranium isotopic ratios or abundances and not the concentration. The last parameter is not generally considered to be of so much interest as the element is ubiquitous in nature and natural levels in urine vary considerably within a given population.

For producing the uranium mixtures and certifying the uranium isotopic abundances the previous methodology used for NUSIMEP 2 and 3 was followed. A matrix was prepared with certifiably low amounts of (natural) uranium; the uranium bulk material was prepared by mixing certified UF_6 and converting it into the nitrate form. The $n(^{235}U)/n(^{238}U)$ was certified by measurement on the MAT 511 gas-source mass-spectrometer and the minor isotope ratios $n(^{234}U)/n(^{238}U)$ and $n(^{236}U)/n(^{238}U)$ were measured and certified by Thermal Ionisation Mass Spectrometry, TIMS using the certified $n(^{235}U)/n(^{238}U)$ values for calibration of the internal mass-fractionation.

As part of general campaigns for measurement of uranium isotopic ratios, eight uranium isotopic mixtures were made as UF_6 and certified by this method. These materials will be used for future CCQM and NUSIMEP campaigns.

Two of the uranium isotopic mixtures were selected for this campaign, one depleted and one slightly enriched.

2. Sample preparation

2.1. Selection of materials

Collection and certification (and keeping over a long time period) of natural urine was not deemed to be a workable route. Instead it was decided to make a 'simulated' urine matrix using the same saline as for NUSIMEP 3 with addition of urea to provide organic material. The saline solution had to be stripped of natural uranium before use in the preparation of the matrix for NUSIMEP 4.

The 'bulk' uranium material was, as outlined above, taken from a series of UF_6 mixtures, and after hydrolysing and converting to the nitrate was dissolved in 1 M nitric acid, thereby creating a solution with 1000 ppm (1 mg $U \cdot g^{-1}$ solution) concentration of uranium. An aliquot of this solution was added directly to the matrix solution before final sampling.

It was also decided to have a higher concentration of uranium in the simulated urine samples than would be expected in 'real' one where typical concentrations of up to 50 ppt (50 pg $U \cdot g^{-1}$ urine) are found. We have no experience of the stability of uranium in such weak solutions and prefer to keep the concentration higher. A concentration of 5 ppb (5 ng $U \cdot g^{-1}$ urine) was chosen following NUSIMEP 3 where such concentrations were found to present measurement difficulties for participants (even though for routine measurements of uranium enrichment in urine considerably lower uranium concentration can be expected).

2.2. Preparation of simulated urine

It was decided to prepare a simulated urine consisting of urea at the concentration of about 17 $g \cdot l^{-1}$ (normal concentration of urea in urine is lower but to simplify the organic composition only urea was used as an organic component) and a saline solution at a similar concentration i.e. about 17 g of salts$\cdot l^{-1}$ in 0.5 M nitric acid.

The urea used for the urine preparation was from Merck (catalogue no.1.08488), chosen out of 4 products of different suppliers considered for this campaign. The decision was taken based on analytical certificates. The saline solution was prepared from ready mixture of inorganic salts (Aquarium Systems, Instant Ocean®), nitrate and phosphate free. Deionised water produced by Milli-Q system (Millipore, USA) was used for dissolving the salts and the urine acidity was adjusted with J.T Baker ultra-pure nitric acid.

All steps of the urine preparation were performed either in the medium-clean (MCL) or ultra-clean chemical laboratories (UCCL) at IRMM. The lab-ware was cleaned following the requirements of trace elements analyses (leaching with acid baths followed by de-acidifying in Milli-Q water). An air survey of working spaces in the laboratories was performed to estimate the potential contamination with natural uranium. Prior to preparation of the matrix solution

each component was analysed for its natural uranium content. As the result of these analyses the saline material was found to contain several ppb of natural uranium and therefore had to be purified from uranium before being used as the basis of the matrix for this campaign. The level of uranium concentration (a few ppt) in urea was considered to be too low to have significant influence on uranium isotopic ratio in final mixtures and urea was used for synthetic urine preparation without purification treatment.

Table 1: Content of simulated urine

	saline components [$g \cdot l^{-1}$]	organic components [$g \cdot l^{-1}$]
Na	5.40	
K	0.20	
Mg	0.65	
Ca	0.20	
Cl	9.60	
SO_4	1.35	
Si	Negligible	
Urea		17.0
TOTAL	17.4	17.0

The saline purification was performed by adjusting the solution to 3 M in nitric acid and passing it through U-TEVA columns (100 - 150 µm mesh resin) from Eichrom Technologies, USA [1]. A large volume of saline solution had to be purified as the solution was needed for NUSIMEP 4 as well as for the CCQM–P48 Pilot Study, for production of the reference urine materials and for forthcoming NUSIMEP campaigns. To speed up the purification process a set of parallel columns was built. In designing the set-up care was taken that the saline solution should have no contact with the air during the whole cleaning process. Moreover, to completely eliminate contamination by natural uranium present in the air, the purification process was performed in the UCCL, class 10.

The solution in 3 M nitric acid was found to be sufficient to ensure nearly 100 % (> 99.5 %) uranium retention as proved by subsequent measurements of the uranium concentration performed by ICP-MS.

After purification, the saline solution was diluted with Milli-Q water and adjusted to 0.5 M in nitric acid and the urea then added to complete the simulated urine matrix solution. The final acidity was checked by titration with TRIS (Tris(hydroxymethyl)aminomethane).

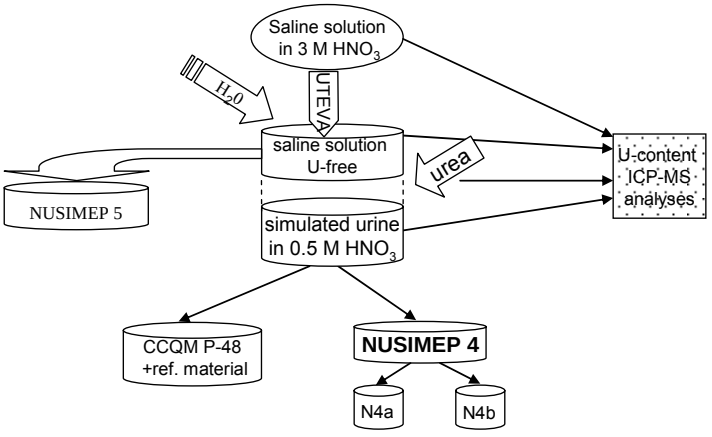

Fig.1. Flow chart of NUSIMEP 4 sample preparation

Both the purified saline solution and the final simulated urine were analysed for uranium traces. The uranium concentration in saline solution was reduced to a value that resulted in a uranium concentration of 5.47 pg·g^{-1} in the final urine solution.

2.3. Production of 'bulk' uranium

As for previous NUSIMEP campaigns uranium isotopic mixtures were prepared in the gas phase by mixing selected, certified uranium material in the form of UF_6. The uranium isotopic ratio $n(^{235}U)/n(^{238}U)$ was then certified by gas-source mass spectrometry, measuring each material relative to two certified UF_6 reference materials. The UF_6 mixtures were homogenised by heating above the triple point and cooling. This was repeated 3 times. Samples of the prepared materials were distilled into small stainless-steel vials, hydrolysed with nitric acid, dried and heated at 300 °C to convert into the UO_3 form and to remove traces of fluoride. The weighed samples of the uranium oxide were converted into uranyl nitrate by dissolving in 8 M nitric acid and evaporating to dryness. The resulting nitrates were dissolved in 3 % HNO_3 to obtain 1 mg U·g^{-1} (1000 ppm) solutions. Aliquots of these solutions were added to the simulated urine to produce N4-a and N4-b solutions. The uranium solutions were also used for subsequent measurements of the minor isotopic abundances by TIMS.

2.4. Mixing 'bulk' and matrix material

To prepare the solutions to be analysed by campaign participants the simulated urine was mixed with the solutions of the uranium isotopic mixtures described above.

Portions of 2 l of the urine were decanted from the common batch into 2 cleaned containers. 10.5 µl of the corresponding solution of uranium isotopic mixture was pipetted into each container to prepare the two NUSIMEP 4 solutions, labelled 'N4-a' and 'N4-b' with 5 ng·g^{-1} (ppb) total uranium concentration.

2.5. Sample containers

Before dispensing, the N4-a and N4-b solutions were stored for about 3 weeks. After bottling into the final containers they were stored overnight in an upside-down position to check that the bottles were leak-tight. The neck was wrapped with Parafilm inside and outside the screw cap. No container showed a leak in this test.

For bottling and transporting the sample solutions to the participants, screw cap polypropylene, plasma cleaned Wheaton (Wheaton Science Products, USA) sampling containers were used.

3. Measurements and certification of uranium

3.1. Natural uranium in the matrix components

The content of natural uranium in all urine components, prior to mixing with the uranium isotopic mixtures, as well as from possible leaching of uranium from the sample containers was measured by inductively coupled plasma mass spectrometry (ICP-MS).

3.1.1. Measurements by Inductively-Couple-Plasma Mass Spectrometry (ICP-MS)

The uranium contents in urea and in the cleaned saline were measured using an Elan 6000 quadrupole ICP-MS (Perkin-Elmer Sciex); the uranium content in the final urine matrix by a single-detector double-focussing magnetic sector Element 2 (Thermo Finnigan). The uranium content in the air survey samples and in solutions used to check the leaching effect in the sample containers were also measured with the Elan 6000. More details about these measurements are published [2].

3.2. 'Bulk' uranium certification by TIMS measurements

The measurement by TIMS has been described previously [3, 4]. The measured value of the $n(^{235}U)/n(^{238}U)$ ratio compared to the value certified by gas-source mass-spectrometry is used to calculate the mass-fractionation factor for each individual measured filament. The $n(^{234}U)/n(^{238}U)$ ratio was measured with both masses in Faraday cups in parallel. The $n(^{236}U)/n(^{238}U)$ ratio was measured with mass 236 by the secondary electron multiplier (SEM) ion counter and the 238 mass on the Faraday cup. The response of the SEM was calibrated by measuring a suitable peak (mass 234 or 235) on both Faraday collector and SEM. The method has been previously described in detail [3, 4] and forms the basis of the measurement of minor isotope abundances of uranium at IRMM.

3.3. TIMS verification measurements of isotopic ratios in samples N4-a and N4-b

The uranium isotopic ratios in the simulated urine mixtures were measured by thermal ionisation mass-spectrometry.

Uranium was separated from the analysed solutions of mixture N4-a and N4-b by absorbing it on U-TEVA resins.

To study if the matrix pre-treatment prior to the recovery of uranium on the resin has an influence on the measurement of the isotopic ratios, samples solutions were treated in 3 different ways before passing through U-TEVA resin. The samples were mineralised:

1. by microwave digestion,
2. by evaporation with a mixture of conc. HNO_3 and H_2O_2,
3. or not mineralised at all.

All solutions were adjusted to 3 M HNO_3 and passed through U-TEVA columns to absorb the uranium. The columns were washed with 20 column free volumes (cfv) of 6 M HCl. Uranium was then eluted from the columns with 10 cfv of 0.01 M HCl. The effluents were evaporated to dryness with the addition of a small amount of 6 M HNO_3, the dry residues were re-dissolved in 0.2 ml of H_2O_2 and again evaporated to dryness to remove the organic traces. The dry residues were then dissolved in 20.0 µl of 6 M HNO_3 and divided into two parts: half of the solution was used for TIMS measurements and half was transferred into 10 ml of 2 % HNO_3 for future ICP-MS measurements.

No influence of the treatment methods could be seen in the TIMS measurements of the uranium isotopic ratios in the samples.

4. Certification of NUSIMEP 4 samples

4.1. Basis of calculation of the final isotopic ratios

The final uranium isotopic ratios of the two NUSIMEP 4 samples were certified based on the certificates of the bulk uranium material, prepared initially as UF_6 with corrections for the measured content of the natural uranium in the matrix materials.

Of the possible contributions to the final uranium isotopic ratios, only the contribution from the natural uranium remaining after purification in the saline solution used to prepare the simulated urine samples needed to be taken into account. The possible contributions from lab-ware and from other sources were demonstrated to be negligible. A comprehensive study was made to check and eliminate such possible sources of uranium. Feasible sources were considered to be reagents, lab-ware and air-borne contamination. Use of cleaned glass-ware (quartz) and plastic-ware, highest (certified) quality ultra-pure reagents, Milli-Q water as well as carrying out all preparatory work in ultra clean chemical environment (UCCL class 10) kept all these contributions of natural uranium effectively to zero levels.

5. Conclusions

The NUSIMEP 4 campaign successfully showed that 'realistic' samples of uranium with certified isotopic content could be produced in a suitable form to test the ability of laboratories to measure uranium isotopic ratios in materials with a high salt and organic content ('simulated urine' in this case). A high proportion of the results have been received up to now and the evaluation process is on-going. This will finish with a Participant's Report to be sent to all participants later in the year.

A parallel CCQM (Comité Consultatif pour la Quantité de Matière) campaign, organized by IRMM, is presently being held on the measurement of uranium isotopes in very similar solutions. A comparison of the results from these two parallel campaigns will be made in the near future.

References:

1. HORWITZ E. P. et al, Separation and preconcentration of uranium from acidic media by extraction chromatrography; Analytica Chimica Acta, Vol.226, 25-37, 1992 (HP392)
2. Quetel C. et al, Report on simulated urine isotopic reference materials preparation and its certification, in preparation
3. Richter S., Goldberg S. A. (2003), Improved Techniques for High Accuracy Isotope Ratio Measurements of Nuclear Materials using Thermal Ionization Mass Spectrometry, International Journal of Mass Spectrometry, 229 (2003) 181-197
4. Richter S., Alonso A., Kühn H., Wellum R., New Procedures for Uranium Isotope Ratio Measurements using the new TRITON Thermal Ionisation Mass Spectrometer, Internal report GE/R/IM/19/04, July 2004

Improvements in Precision and Accuracy of Uranium Isotope Ratio Measurements for Reference Material Certification at IRMM

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Keywords: TIMS, REIMEP, Uranium isotopes

Abstract:

For the certification of nuclear isotopic reference materials a new state-of-the-art thermal-ionisation mass-spectrometer (TIMS), the Thermo Electron "Triton", was installed at IRMM in January 2004. Significant progress has been made particularly for measurements of the $^{234}U/^{238}U$ and $^{236}U/^{238}U$ (the so-called "minor") isotope ratios.

Using new measurement procedures for the Triton mass spectrometer, more precise and accurate isotopic data for the minor isotopic ratios have been obtained for a number of old and new reference materials, thus allowing a (re-) certification of their isotopic composition. This includes the well known IRMM-183-187 series and also the REIMEP 15 campaign samples. During the re-certification of the REIMEP 15 samples, the observed deviation between the participant results and the previously certified ratios by IRMM in certain cases has been overcome.

1. Progress in measuring minor uranium isotope ratio measurements

The variation of the uranium isotopic composition between samples of different origin as well as the dynamic range of the isotope ratios is large compared to most other elements. The naturally occurring isotopes ^{235}U and ^{238}U are usually considered as the major isotopes whereas ^{234}U and ^{236}U are called minor isotopes because they are significantly less abundant (factor 10^{-2} to 10^{-10}) in most samples.

The variety of sample compositions and especially the extreme dynamic range of the ratios place considerable demands on mass spectrometers, in particular the need for high abundance sensitivity and ionization efficiency, low background and noise, linearity and high efficiency of detectors. Therefore accurate measurements of the isotopic ratios for uranium samples have always been a challenge for the analyst. Over the last 20 years a significant improvement in the uncertainties for minor uranium isotope ratios has been achieved. This is shown in Figure 1.

Starting with the uncertainties given on the original certificates of IRMM 183-187, the uncertainties were first improved by a factor of about 10 using the MAT262, a previous generation TIMS, in Faraday mode ("MAT262-Faraday") as well as in SEM (Secondary Electron Multiplier) ion counting mode ("MAT262-SEM") applying the ^{235}U beam for the inter-calibration between Faraday cups and the SEM. The uncertainties were improved by a factor of 10 again using a significantly higher ^{238}U ion beam only possible on the Triton, both in Faraday mode ("Triton-Faraday") and in SEM ion counting mode ("Triton-SEM") with ^{234}U as the calibrating beam.

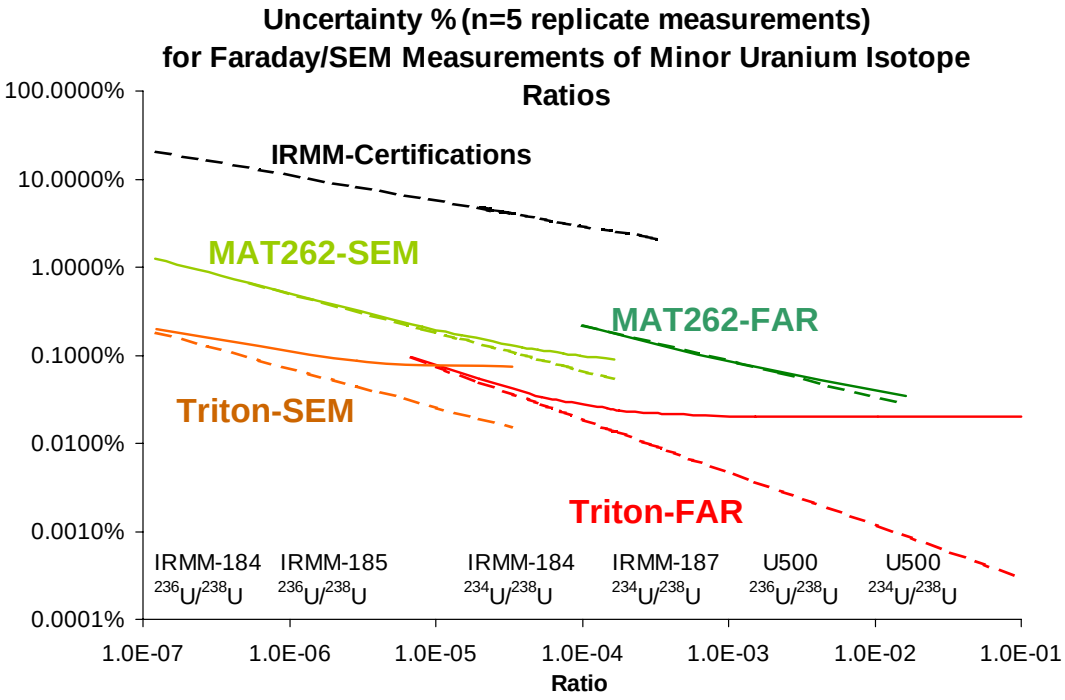

Fig. 1: Uncertainties for $^{234}U/^{238}U$ and $^{236}U/^{238}U$ measurements using the Triton TIMS, compared with the MAT262 TIMS, and the uncertainties of earlier certified values by IRMM. All solid lines include the uncertainty contribution from the mass fractionation correction (internally using the $^{235}U/^{238}U$ ratio); the dotted lines do not include this contribution.

New data measured on the of the well known IRMM-183-187 series have been used for to recertify their minor isotope ratios with much smaller uncertainties:

	$^{234}U/^{238}U$	$^{235}U/^{238}U$	$^{236}U/^{238}U$
IRMM-183	$1.9755(22)\cdot10^{-5}$	0.003 215 7(16)	$1.48358(54)\cdot10^{-4}$
IRMM-184	$5.3138(32)\cdot10^{-5}$	0.007 262 3(22)	$1.2446(17)\ 10^{-7}$
IRMM-185	$1.79474(80)\cdot10^{-4}$	0.020 055 2(60)	$2.8889(23)\ 10^{-6}$
IRMM-186	$2.9365(13)\ 10^{-4}$	0.030 771 1(92)	$3.3219(23)\ 10^{-5}$
IRMM-187	$3.8700(16)\cdot10^{-4}$	0.047 325(14)	$7.1965(39)\ 10^{-5}$

Table 1: Re-certification of the IRMM 183-187 series of reference materials. The new certificates, including the isotopic abundances and mass fractions can be obtained from IRMM. Uncertainties are calculated according to the GUM [1].

2. Review of REIMEP 15 Campaign

The REIMEP (Regular European Interlaboratory Measurement Evaluation Programme) campaign 15 for isotopic ratios of uranium in UF_6 samples was completed in 2001. The participants received 4 samples of low-enriched or slightly depleted uranium in the form of UF_6 in standard Monel capsules. The samples were prepared at IRMM by mixing uranium reference materials certified by gas mass-spectrometry in the form of UF_6. The minor isotopes, ^{234}U and ^{236}U were certified at IRMM by Thermal Isotope Mass-spectrometry (TIMS) using a MAT262-RPQ mass spectrometer in ion counting mode. For the isotope ratio $n(^{234}U)/n(^{238}U)$ the comparison of IRMM's certified values for at least one of the samples with the participant's results showed a significant relative deviation of about 2%, which is considerably larger than the typical uncertainties expected for this kind of measurements. This problem has now been overcome firstly by measurements using the new TRITON TIMS at IRMM and secondly by a thorough investigation into the reasons for the deviation observed in the past.

Because of the extended dynamic range of the new Faraday multi-collector all $n(^{234}U)/n(^{238}U)$ and $n(^{236}U)/n(^{238}U)$ ratios (except $n(^{236}U)/n(^{238}U)$ for sample C) can be measured using Faraday detectors alone. On the TRITON even the "minor" isotopes ^{234}U and ^{236}U could be measured for most of the samples using Faraday detectors, whereas previously with the MAT262-RPQ ion counters had to be used. By measuring only with Faraday detectors the inter-calibration between Faraday cups and the ion counter as well as the linearity and the dead time corrections on the ion counter are all avoided. This leads to lower uncertainties on the measured ratios.

In Figure 2 the results for the sample that showed the greatest difference between the previous IRMM certified value for the minor isotope ratio $n(^{234}U)/n(^{238}U)$ and the values measured by participating laboratories are shown together with new results obtained recently at IRMM. The TRITON result now agrees very well with the average value from all the participating laboratories. A detailed description of the re-measurements for REIMEP 15 has been published [2].

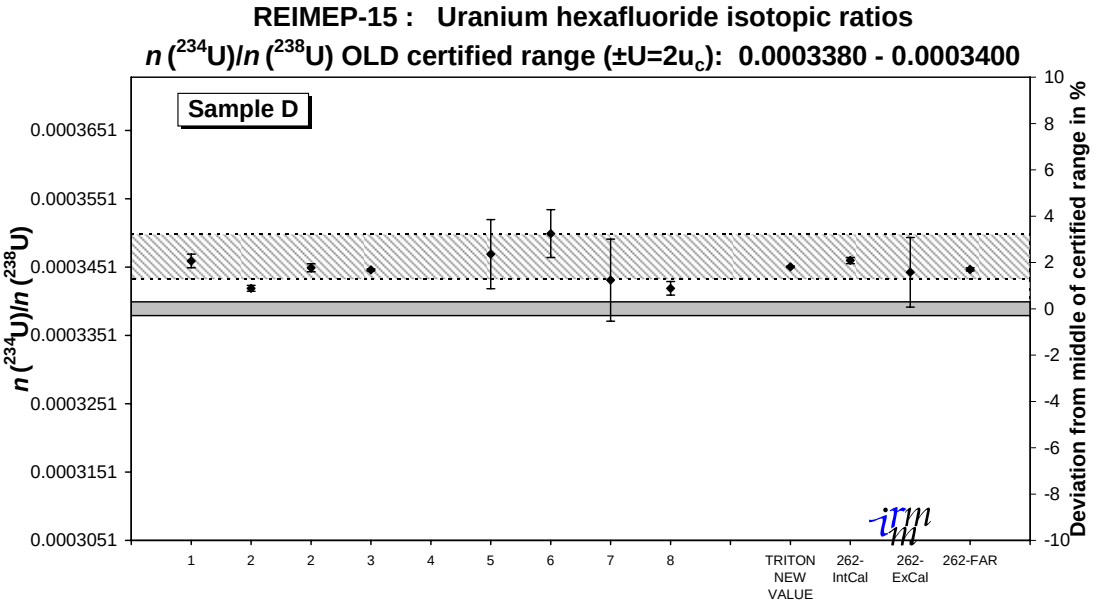

Fig. 2: Results for $n(^{234}U)/n(^{238}U)$ of REIMEP 15 D from all participants (1-8), together with recent re-measurements at IRMM. The indicated "TRITON NEW VALUE" is used for the re-certification.

3. Conclusions

Advances in measurement techniques associated with the improvements in the new generation of TIMS instruments has allowed us to carry out remeasurements on some reference materials and also to recertify the uranium isotopic ratios of the minor isotopes in REIMEP 15 where some discrepancies between certified values and participants' measurement results had been previously noted.

Building on these results and the experience gained we plan to prepare and certify isotopic RMs concentrated on certifying the minor isotopes in particular.

References

[1] ISO Guide to the Expression of Uncertainty in Measurement, 1st edition, 1993, corrected and reprinted 1995, International Organisation for Standardisation (Geneva, Switzerland)

[2] Update of REIMEP-15: Isotopic ratios of Uranium in UF6, S. Richter, A. Alonso, W. De Bolle, H. Kuehn, A. Verbruggen, R. Wellum, EUR Report 21562 EN

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Verification of MTR Fuel Plates by Gamma Spectroscopy

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Abstract:
The verification of the content of ^{235}U and the verification of the active length of typical MTR fuel plates is described. Only gamma spectroscopy and direct calibration with standards are used. The measurements are performed scanning the fuel plate axis with a collimated detector. The 185.7 keV photon net peak area of the gamma spectrum is used to determine the ^{235}U content. The activity measured at different axial positions of the fuel plate verifies the homogeneity level of the sample. The active length of the fuel plate is calculated with an iterative weighted least square fit of the activity profile using a well-known function.

Keywords: gamma scanning, nuclear safeguards, NDA

1. Introduction

The paper describes the performance of two gamma-scanning systems designed for the verification of MTR (Material Test Reactor) fuel plates. The MTR Gamma Scanner has been developed for the Inspection Directorate of DGTREN for the verification of MTR fuel assemblies in a fuel fabrication plant. The system is equipped with a germanium n-type coaxial detector and an automatic scanning device. The second gamma scanning system installed in the NDA laboratory of the Nuclear Safeguards unit is equipped with a NaI detector, a linear drive unit and acquisition electronics.
Both Gamma Scanner systems assay the mass of ^{235}U in the fuel plates by application of gamma spectroscopy. The ^{235}U mass is proportional to the integral of the net peak areas of the 185.72 keV photon peak of the measured activity profile of fuel plate. Determination of the peak area is very much simpler when using a germanium rather than a NaI detector.

A MTR fuel plate is a “sandwich” of the “active” part (fuel alloy or meat) and the cladding. The “meat” (~0.5mm thick) is usually a mixture of uranium and aluminium or uranium and silicon. The cladding material is aluminium in most cases. The overall length of an MTR fuel plate is commonly less than 1m. The enrichment of MTR fuel ranges from <20% to 93%. The meat thickness of a single plate is always less than the infinitive “thickness”.

2. Description of the gamma scanner systems

2.1. MTR Gamma Scanner

Fig. 1: MTR Gamma Scanner

The detection system of the MTR gamma scanner is composed of a High Purity Coaxial Germanium Detector (ORTEC HPGe GEM15180) with built in preamplifier and the Ortec DSPEC Digital Gamma-Ray Spectrometer where the pulse shaping is accomplished using Digital Signal Processing (DSP). The detector is housed in a combination of collimator/shield, which isolates the detector from background radiation, while providing an accurately collimated window allowing the detector to see only the segment of

the fuel plate being measured. The height adjustable scanner bench structure carries the detector subsystem, the linear drive and the scanning table. The scanning table for positioning the fuel element in front of the detector is powered by a servo positioning system and designed for fuel plates with an active length up to 1.10 m. The Gamma Scanner analysis software determines the activity profile related to the distribution of the ^{235}U inside the fuel plate (Figure 2).

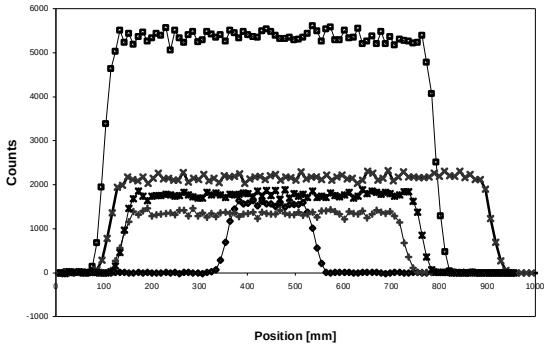

Fig. 2: Scanning profiles obtained with a germanium detector of fuel plates with different active length, mass and enrichment (Measurement Time: 900 seconds, Channels: 90).

The mass of ^{235}U of the fuel plate is proportional to the integral of the net peak areas of the 185.7 keV peak. The net peak area is equal to the total numbers of counts in a region of interest (ROI) minus the background, which can be estimated on the assumption of a straight line.

2.2. NDA Laboratory Gamma Scanner

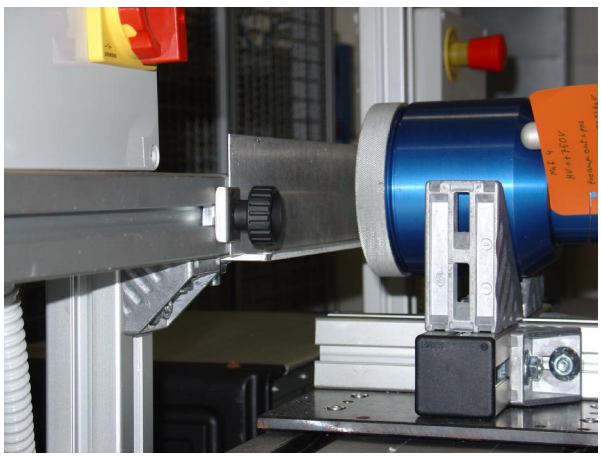

Fig. 2: NDA Laboratory Gamma Scanner

The detection system of the gamma scanner installed in the NDA laboratory is composed of a NaI detector coupled with a photomultiplier and the MCA-166 Gamma-Ray Spectrometer. The detector is housed in a combination of collimator/shield, which isolates the detector from background radiation, while providing an accurately collimated window allowing the detector to see only the segment of the fuel plate being measured. The scanner bench structure carries the holder for the fuel plate and the Isel linear drive with the detector. The RADAR (Remote Acquisition of Data and Review) Data Acquisition Modules *LeuGsDAM* configures the MCA-166 and the Isel Motor Drive, controls the scanning of the fuel plate and writes the activity profile (Figure 3) to a binary data file.

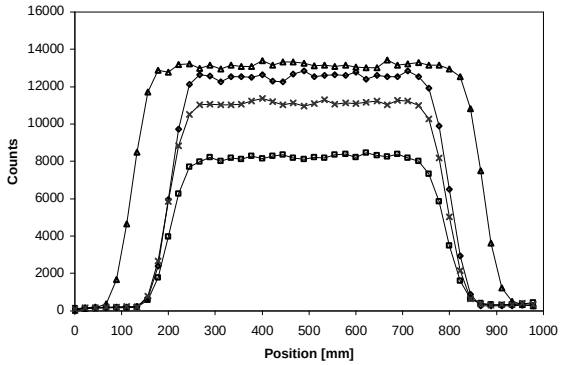

Fig. 3: Scanning profiles obtained with a NaI detector of fuel plates with different active length, mass and enrichment (Measurement Time: 900 seconds, Channels: 90).

The mass of ^{235}U of the fuel plate is proportional to the integral of the net peak areas of the 185.7 keV peak, which in the case of a NaI detector contains all the ^{235}U peaks from 143 keV to 205 keV. The background cannot be considered as a straight line and must be measured in a region of interest (ROI_2) outside of the peak. It is stated that the background under the peak is proportional to that in ROI_2.

3. Data Treatment

3.1. Active Length

The active length of the fuel assembly is obtained by fitting with an iterative weighted least square fit the scanning profile (Figure 4) from the 186 keV net peak areas using the equations (1) and (2).

R(x) represented using the arctangent function:

$$R(x) = \frac{R_P}{p}\left(arctg\left[\frac{x - x_0}{D_1}\right] - arctg\left[\frac{x - (x_0 + L)}{D_2}\right]\right) + B \quad (1)$$

R(x) represented using the error function:

$$R(x) = \frac{R_p}{2}\left(erf\left[\frac{x - x_0}{D_1}\right] - erf\left[\frac{x - (x_0 + L)}{D_2}\right]\right) + B \qquad (2)$$

where π or ½ are the normalization constants for returning the plateau count rate $R_{p.}$ D_1 and D_2 are the length resolution parameters of the system and x_0 is the bottom edge of the source. D_1 and D_2 depends on the detector-source distance and the collimator geometry. The iterative weighted least square fit of the axial profile results in the bottom edge x_0 and in the active length of the fuel assembly (Figure 3).

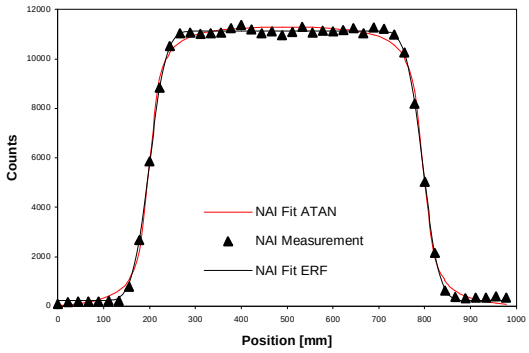

Fig. 4: Scanning profiles obtained with a NaI detector fitted with arctangent function and error function.

3.2. Determination of ^{235}U mass and Calibration constant

Considering the whole fuel plate, the ^{235}U total mass is supposed to be proportional to the sum of the counts from each elementary axial segment. Therefore the ^{235}U mass is computed by integrating the net peak area of the 185.72-keV for the germanium detector or the 143 keV to 205 keV region for the NaI detector over the entire scanning profile (including the tails below x_0 and above x_0+L):

$$M_{235} = K_{235} \sum_{p=1}^{N_p} Counts_{235\,p} \qquad (3)$$

where K_{235} is a calibration constant.

The calibration constant K_{235} is obtained by inverting the equation (3) using standard samples with known mass and enrichment. Due to the linearity of the equation, the calibration constant can be determined with a single standard. Anyway the use of several standards of different mass and enrichment is recommended.

4. Measurement campaigns

Two measurement campaigns of standard fuel plates are reported. The first campaign is in laboratory conditions with a NaI-detector and the other campaign consists in an in-field inspection using a germanium detector in a fuel fabrication plant

4.1. PERLA laboratory campaign

The measurements in laboratory conditions were performed in PERLA. PERLA is a unique facility within the EU that houses an extensive collection of well-characterized nuclear reference material. In the reported campaign the acquisition time used for each scan is 900s and 20s for each channel steps of 22.225 mm. The calibration constants are determined using representative standard fuel plates with different enrichments (Figure 5).

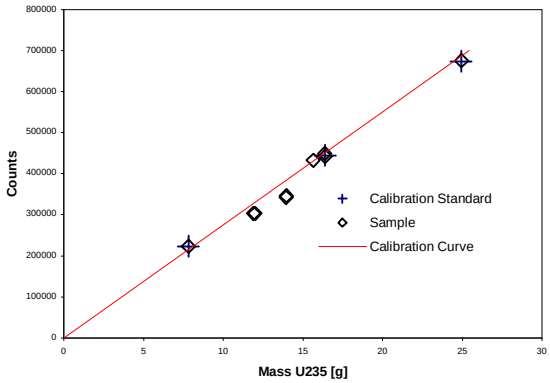

Fig. 5: Calibration curve with standard and sample measurements.

In annex A, Table A reports respectively the ^{235}U mass with the associated uncertainties, for both declared and measured values. The difference between the measured and the declared values (d_{M-D}) with the corresponding uncertainty (s_d) is also shown. This form in presenting the results allows the inspector to check if the condition d_{M-D}<3*s_d is respected.

4.2. Fuel fabrication plant campaign

The campaign was performed in a European fuel fabrication plant during a routine inspection (PIV). The total acquisition time was 900s with 10s measurements per channel. The fuel plates were moving with constant speed. The calibration constants were determined using standards available in the fabrication plant. The standards cover the mass and the enrichment range of the fuel plates to be assayed (Figure 6).

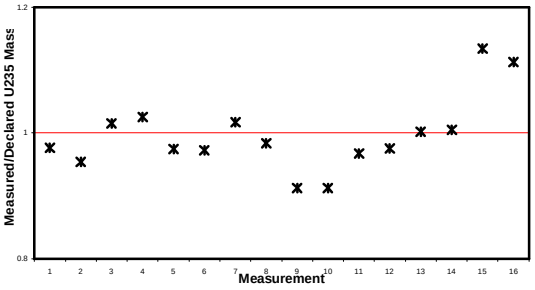

Fig. 6: Ratio of measured and declared ^{235}U mass for fuel plates

In annex A, the Table B reports the ratio between the declared and measured ^{235}U mass with the associated uncertainties and the measured active length.

5. Conclusions

The proposed method for the verification of the U^{235} mass and the active length of MTR fuel plates with gamma spectroscopy has been discussed and applied. The method allows the verification of U^{235} mass and active length of MTR fuel plates with different enrichment and different geometry using either a gamma scanner system with a germanium or a NaI detector. The accuracy of a few percents is sufficient for most inspection requirements.

The Nai based system has the advantage of higher efficiency, low cost, transportability and no need of liquid nitrogen cooling. Integrated into the RADAR/CRISP environment the gamma scanner system could work unattended.

8. References

[1] A. Ravazzani, , R. Jaime, M. Looman, P. Peerani, U. Weng, P. Schillebeeckx and A. Foglio Para; *A Segmented Gamma Scanner for the Verification of LEU Oxide Powders*, Journal of Nuclear Materials Management, Winter 2005 Volume XXXIII, Number 2, p10

ANNEX A

Sample	Declared					Measured			d_{M-D}	
	U^{235} (g)	U_{tot} (g)	E%	err	%	U^{235} (g)	(±)	(%)	d_{M-D}	$d_{(M-D)}$%
pl22	11.88	60.87	19.52	0.30	2.5	10.99	0.82	7.46	0.88	7.43
pl98	13.94	70.1	19.89	0.28	2.0	12.41	0.93	7.49	1.52	10.93
pl56_2	16.38	36.59	44.77	0.29	1.8	16.06	1.2	7.47	0.32	1.98
pl57	16.36	36.56	44.75	0.29	1.8	16.29	1.22	7.49	0.07	0.41
pl94	13.95	70.49	19.79	0.28	2.0	12.58	0.94	7.47	1.37	9.82
pl25	11.98	60.95	19.66	0.30	2.5	11.01	0.82	7.45	0.98	8.14
pl88	7.82	8.4	93.10	0.18	2.3	8.08	0.6	7.42	-0.26	-3.37
pl8588	15.64	16.8	93.10	0.36	2.3	15.70	1.17	7.45	-0.06	-0.39
ml36	24.93	26.76	93.16	0.50	2.0	24.50	1.83	7.47	0.43	1.71

Table A: Declared and measured ^{235}U mass for PERLA samples. d_{M-D} is the difference between the measured and the declared values. In the last column 100(M-D)/D represents, in percentage, the ratio between d_{M-D} and the declared value.

Sample	Declared	Active Length			Mass U^{235}		
	S_d	D	M	d_{M-D}	$d_{(M-D)}$%	sd	Diff %
1-1	0.024	613	624.67	-11.67	0.976476	0.036	-2.352
1-2	0.024	613	648.85	-35.85	0.953913	0.036	-4.609
2-1	0.020	600	590.94	9.06	1.015714	0.037	1.571
2-2	0.020	600	590.64	9.36	1.025714	0.036	2.571
3-1	0.020	900	908.03	-8.03	0.974857	0.036	-2.514
3-2	0.020	900	907.92	-7.92	0.972571	0.036	-2.743
4-1	0.010	700	693.09	6.91	1.016939	0.036	1.694
4-2	0.010	700	694.12	5.88	0.98411	0.036	-1.589
5-1	0.011	800	798.22	1.78	0.912745	0.037	-8.725
5-2	0.011	800	793.27	6.73	0.913072	0.037	-8.693
6-1	0.018	750	743.88	6.12	0.967972	0.036	-3.203
6-2	0.018	750	741.74	8.26	0.975534	0.036	-2.447
4-3	0.010	700	691.37	8.63	1.002099	0.036	0.210
4-4	0.010	700	693.67	6.33	1.005247	0.036	0.525
7-1	0.019	186	184.62	1.39	1.134043	0.036	13.404
7-2	0.019	186	184.66	1.34	1.112766	0.036	11.277

Table B: Ratio between measured and declared ^{235}U mass for the MTR fuel plates and the corresponding uncertainties for fuel fabrication plant samples

Development of Safeguards System for Lab-Scale ACP Facility Based on the DUPIC Safeguards Technology

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Abstract:

The Korea Atomic Energy Research Institute (KAERI) has been developing the DUPIC (Direst Use of PER spent fuel in CANDU) fuel cycle and the ACP (Advanced Spent Fuel Conditioning Process) technology for the purpose of spent fuel management. A safeguards system has been applied to the R&D process for fabricating DUPIC fuel directly with PWR spent fuel material. The lab-scale DUPIC facility safeguards system was successfully established under the international cooperation program.
ACP is an electro-metallurgical treatment technique to convert oxide-type spent nuclear fuel into a metallic form. This electrolytic reduction (ER) concept was based on the original pyro-processing technology, and was simplified by conducting the lithium electro-winning and reduction process at uranium cathode simultaneously. Based on the DUPIC safeguards implementation experience gained from the DUPIC facility, a safeguards system for the lab-scale ACP facility was designed.
The ACP safeguards system has many unique design specifications because of the particular characteristics of process materials and restrictions in facility operation. For the material accounting system, a set of remote operation and maintenance concept has been introduced for a non-destructive assay (NDA) system. In addition to the NDA system, an intelligent surveillance system based on artificial neural intelligence and the iPIX technology has been developed for the unattended process monitoring. This paper summarizes the main features and the current status on developing the ACP safeguards system with emphasis on the intelligent surveillance system. It is expected that deployment of these advanced surveillance technologies would be useful for construction of the robust safeguards system for the lab-scale ACP facility.

Keywords: DUPIC, ACP, Spent fuel, NDA, Surveillance

1. Introduction

DUPIC fuel cycle concepts were developed in consideration of reutilization of over-flowing spent fuel resources at PWR sites and reduction of high level wastes generated. DUPIC was first introduced at the meeting of ROK-US Joint Standing Committee on Nuclear Energy Cooperation (JSCNEC) in early 1991 in an effort of looking into a fuel cycle technology that would not be implicated with international nonproliferation policy in using plutonium bearing spent fuel material. Subsequently with a favorable international consensus DUPIC project could go on with the feasibility study on several options of DUPIC process and its safeguardability. Safeguards responsibility at this point then was to create a total system that satisfy obligations required by both US and IAEA in having access to handling of

spent fuel and measurement of bulk form of spent fuel. As for US obligations DUPIC project needed to produce justifiable source information and data in a form of report to satisfy US for leading to ROK-US Joint Determination, commonly known as prior consent clause of alteration in forms and contents of irradiated US origin fuel. In the case of IAEA, DUPIC facility was necessary to come up with the viable measurement system for process material accounting, and to provide a means of maintaining continuity of knowledge of process material flow, with which IAEA could develop its own facility specific DUPIC safeguards approach. After about 10 years of technology development and closer international cooperation with US, AECL and IAEA, DUPIC project has finally come to the point where hot material was introduced in the DFDF(DUPIC Fuel Development Facility) hot cell and has been in normal use.

The advanced spent fuel conditioning process (ACP) has been under development at KAERI since 1997 to tackle the problem of the accumulation of the spent fuel. The concept is to convert the spent oxide fuel into a metallic form in a high temperature molten salt in order to reduce the heat power, volume, and radioactivity of the spent fuel. The main objective of the ACP is to treat the PWR spent fuel for a long-term storage and eventual disposal in a proliferation resistant and cost effective way. Moreover, the electrolytic reduction method of the ACP can contribute to the innovative nuclear energy system as a key technology for the preparation of the metallic fuel. Since the inactive tests of the ACP have been successfully implemented to confirm the validity of the electrolytic reduction technology, a lab-scale hot test will be undertaken in a couple of years to validate the concept. A preliminary study on the safeguardability of a pilot-scale ACP facility was performed. As a result of the study, our conceptualization of facility features and material flows across the ACP facility lead us to conclude that a safeguards system could be designed to meet the IAEA's detection goals and to provide an independent verification scheme. Based on the results of a safeguards implementation at DFDF hot cell, the reference safeguards design conditions are established for the ACPF.

This paper addresses the main features of DUPIC safeguards development status and also adds future prospects of safeguards implementation to the ACP facility at KAERI.

2. DUPIC Safeguards System Development and Implementation

2.1. Technology Development of DUPIC Safeguards System

Safeguards system in regard to DUPIC fuel cycle faced many issues to cope with, namely: 1) there was no directly applicable NDA measurement technology for facility accounting for spent fuel material due to high radiation and fission product interference, 2) there was no DUPIC specific safeguards criteria available in IAEA reference publication, and 3) there were no appropriate methods as C/S system for the positive identification of material movement in and out of hot cell in shielded cask. DUPIC safeguards R&D group could summarize technologies necessary for process material accounting as follow: (1) neutron coincident counting methodology could be one alternative fabrication facility, (2) Near Real Time Accountability system would be required due to the inaccessibility to process material under high radiation environment, and lastly (3) unattended continuous monitoring system comprising of radiation detection and image recording would be built into the system for maintaining continuity of knowledge of material flow as dual C/S system.

In establishing accountability system to carry out a lab scale of facility aimed to produce sample fuel pins for irradiation in research reactor, there are four different nuclear facilities at KAERI, namely, PWR, PIEF(Post Irradiation Examination Facility), DFDF, HANARO, and RWTF (Radioactive Waste Treatment Facility).

2.2. NDA Measurement Method

As for the accounting method, PWR, PIEF, and reactors are considered to be item counting facilities and DFDF is a bulk measurement facility. Normally item facilities need to account for nuclear material based on burn up data and code calculations. Bulk facility need to account for nuclear material by weighing, chemical analysis of representative samples, and/or NDA measurement. But in the case of DUPIC process line NDA method is adapted to account for entire process material. In order to maintain data coherence, and material balance between item facilities and bulk handling facility and

also to remove Shipper/Receiver Difference occurrence Curium Ratio method is introduced to account for DUPIC process material, in which accounting data from item facility is normalized to NDA measurement data in DFDF provided that NDA measurement system is authenticated by IAEA. Under the Curium Ratio method all of internal measurement on samples is done with one instrument called DSNC (DUPIC Safeguards Neutron Counter).

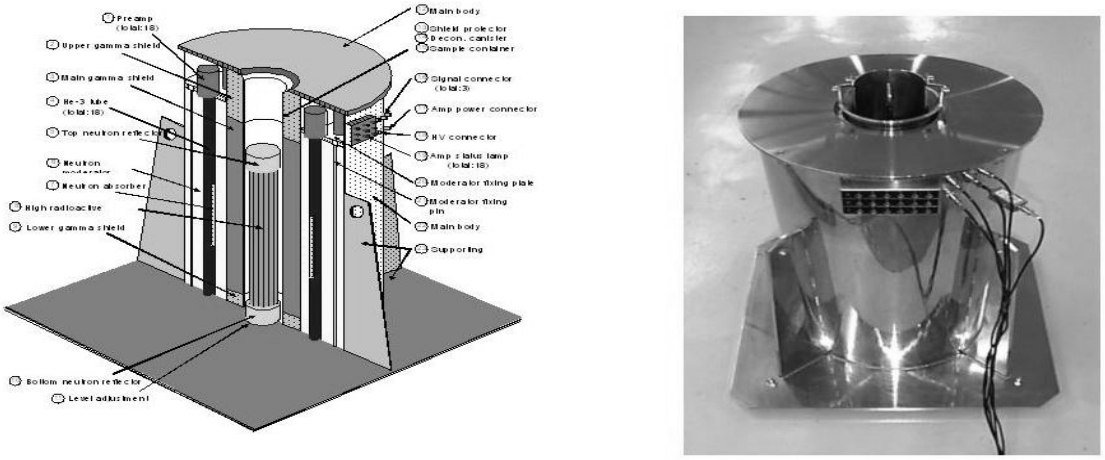

Fig. 1 Drawing and photo of DUPIC Safeguards Neutron Counter

The drawing and the photo of manufactured DSNC are shown in figure 1. The DSNC, a well-type neutron coincidence counter, is for inferring the amount of Cm-244 from measuring spontaneous fission neutrons at various process stages in DFDF. The DSNC design focused on all types of DUPIC process material that are remotely measurable (CANDU type bundle, powder, rod-cut, hulls, and wastes) in a hot cell during lab scale operation. A total of 18 He-3 tubes with nitrogen quenching were symmetrically located in a high-density polyethylene moderator and each of the He-3 tubes was connected to an individual preamplifier to reduce the gamma-ray pileup problem. A preamplifier status lamp was attached to each tube in order to visually monitor its normal operation in the hot cell. Another unique feature of the DSNC, compared to other conventional coincidence counters, is that substantial shielding is added to protect the He-3 tube/electronics from the intense gamma-rays of the DUPIC process material with a maximum surface dose of ~10^4 R/h.

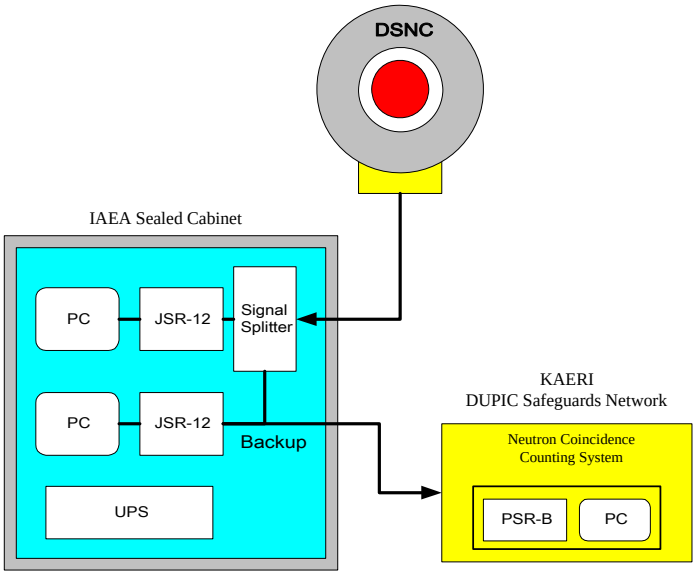

Fig. 2 Connection of the DSNC under IAEA Safeguards

2.3. Material Accounting System

In order to have a better quality of IAEA safeguards the operator side suggested that the construction of process built-in C/S system should be installed from the design stage to facilitate IAEA independent verification activities and also to minimize interference of normal facility operation. For that purpose, facility operators accepted the authentication of process measurement system by installing an IAEA instrument calibration source and allowing the raw signal of the measurement data to be routed directly to an IAEA monitoring station. In addition, operator and IAEA agreed to have either a surveillance camera or seals on all un-monitored openings of the DUPIC facility.

Fig. 2. shows the connection of cabling DSNC as well as IAEA's and KAERI's instruments related to the DSNC. As shown in this figure, the neutron signal from DSNC was split into 3 neutron signals through a splitter that was inside the IAEA cabinet. Two signals from the splitter connected to two JSR-12s of the IAEA system, respectively, of course one of them is for backup, and the remaining one was connected to the PSR-B of the KAERI system. In addition, the IAEA neutron source was located under the VACOSS seal near the DSNC in the M6 hot-cell to determine the detector characteristics in advance of DUPIC material measurement. According to the experimental calibration test, the absolute efficiency the DSNC system was determined as 13.5%.

2.4. Containment and Surveillance System

When structuring a nuclear safeguards system for the DUPIC facility, the secondary concern is the containment and surveillance system. For IAEA safeguards, the construction of process built-in C/S system was installed from the design stage to facilitate IAEA independent verification activities. The agency's surveillance system for DUPIC facility includes five DCM-14 digital cameras, two neutron monitor, and SDIS (Server on Digital Image Surveillance System). In addition to the IAEA system, DUPIC safeguards team also established the their own surveillance system named as DSSS (DUPIC Safeguards Surveillance System) for technology development which is shown in Fig. 3.

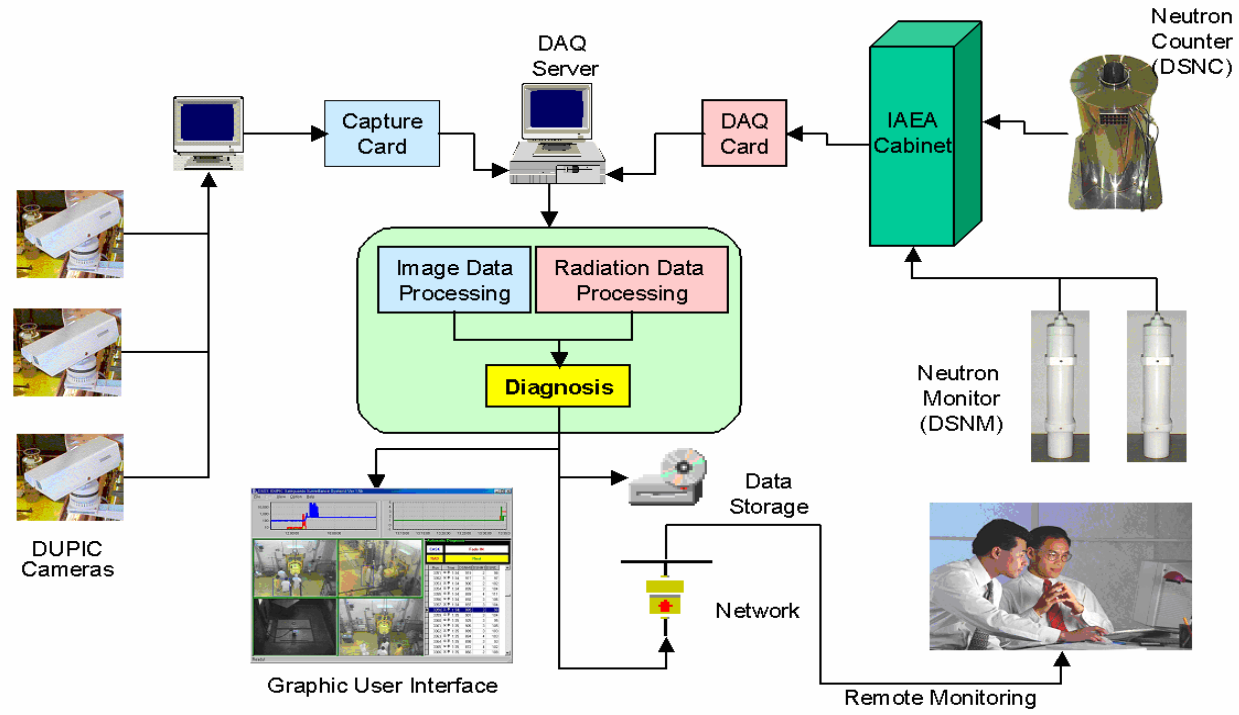

Fig. 3. Configuration of the DUPIC Surveillance System

The R&D efforts in the C/S area are directed to develop an unattended, continuous, integrated surveillance system to meet and improve the basic functionality of other unattended continuous surveillance systems. In the system development, particular effort is made for digital analysis of events

by incorporating an advanced diagnosis mechanism to selectively draw a conclusion on only the significant events throughout the monitoring period. This was done by integrating the video and radiation sensors' data in a common time dimension through image processing and designing a computer interface for neutron counting. This system is able to alert spent fuel material movement to and from a typical hot cell system. The diagnosis routines of both radiation and image determine the transportation status to one of the five cases of 'No Detection', 'Near Field', 'Fade In', 'Rest', and 'Fade Out'. The prototype of the intelligent C/S system using artificial Neural Networks was developed and the performance test of the system has been conducted. According to the results of performance test operations, a series of revision on the C/S system had been conducted including the implementation of remote monitoring function.

3. ACP Safeguards System

Based on the results of a safeguards implementation at DFDF hot cell, the reference safeguards design conditions are established for the ACPF. A conceptual design of safeguards system for ACPF is described in Fig. 4. Basically, the nuclear material accounting will be performed by a well-typed NDA system, so-called ASNC(ACP Safeguards Neutron Counter), which is the same concept as the adopting a curium ratio technique.

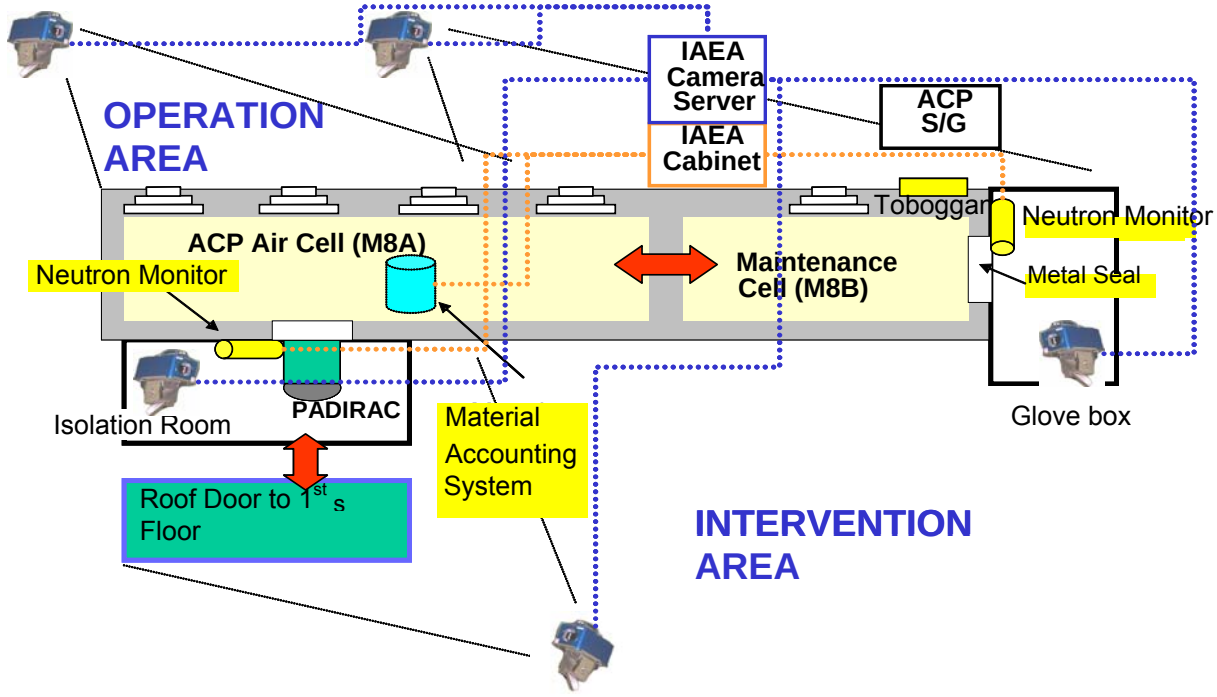

Fig. 4 Conceptual Design of a Safeguards System for the ACPF

Upgrading the DSNC in considering high-efficiency, high-accuracy and remote friendly manner for the hot cell operation has developed the ASNC. The prototype design is shown in Fig. 5. This ASNC was specially designed to fulfill the remote maintenance using manipulators. All connectors of electronic components of the ASNC are Lemo connectors and the geometry of the ASNC is horizontally laid. So the geometry of the ASNC is a little different from any other neutron counters developed by now. This horizontally laid geometry makes it possible to perform easy operation and maintenance in the ACP hot cell. The geometry of components including high density polyethylene, inner and outer gamma-ray shields, top and bottom reflectors, the position of 24 He-3 tubes and cadmium neutron absorbers was calculated by MCNPX code.

There are three types of samples to account in the ACP hot-cell operation; the U metal ingot, the UO2 and U3O8 powder or the rod cut, and the salt waste. From the multiplication analysis, we don't need the triples neutron rate measurement and it is possible to account these samples through singles and doubles rate measurement. Therefore, the SNM(Special Nuclear Material) data of all samples could

be measured by the ASNC as seen Fig. 6. In general, counters that measure only singles and doubles need a flat spatial response and a reasonable detection efficiency (20 ~ 30 %). The mean neutron detection efficiency of the ASNC is about 23.8 % (σ = ±0.7 %) in the axial direction and 23.5 % (σ = ±0.88 %) in the radial direction. The measurement of the parameters of the ASNC is being done and the preliminary result of the efficiency measurement is 21.5 %. 24 PDT's and He-3 tubes are divided into 4 groups, so each group contains 6 PDTs and 6 He-3 tubes, respectively.

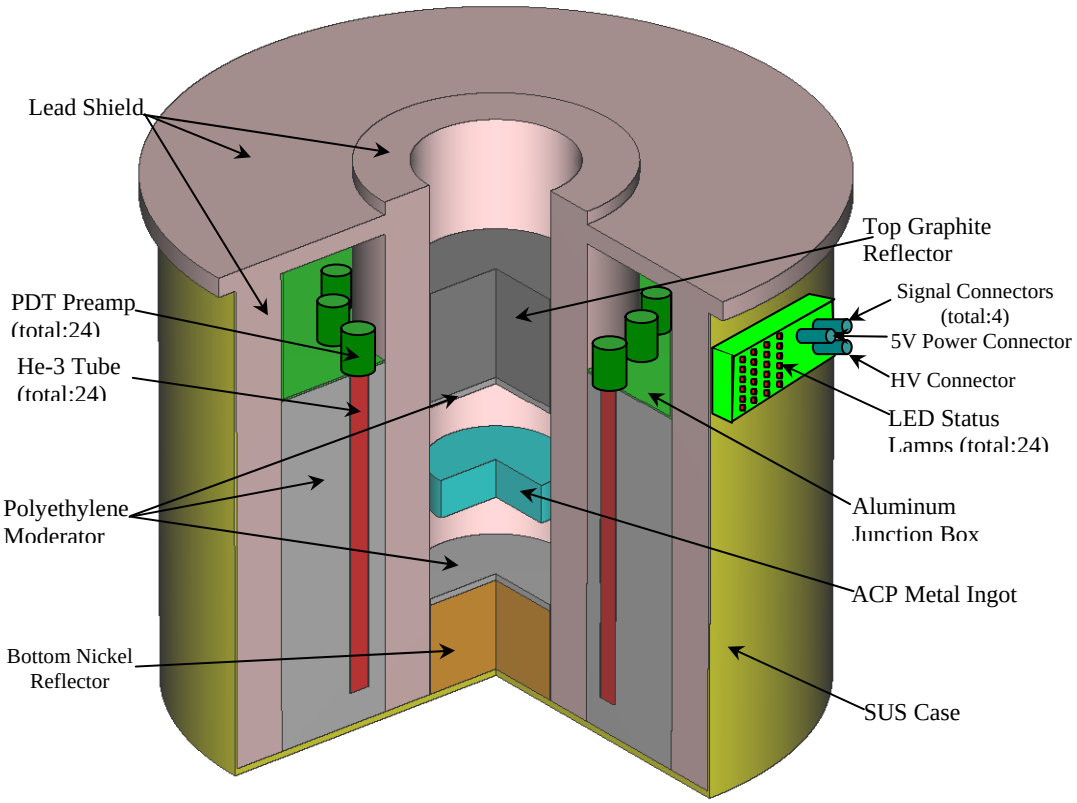

Fig. 5 3D View of ASNC Design

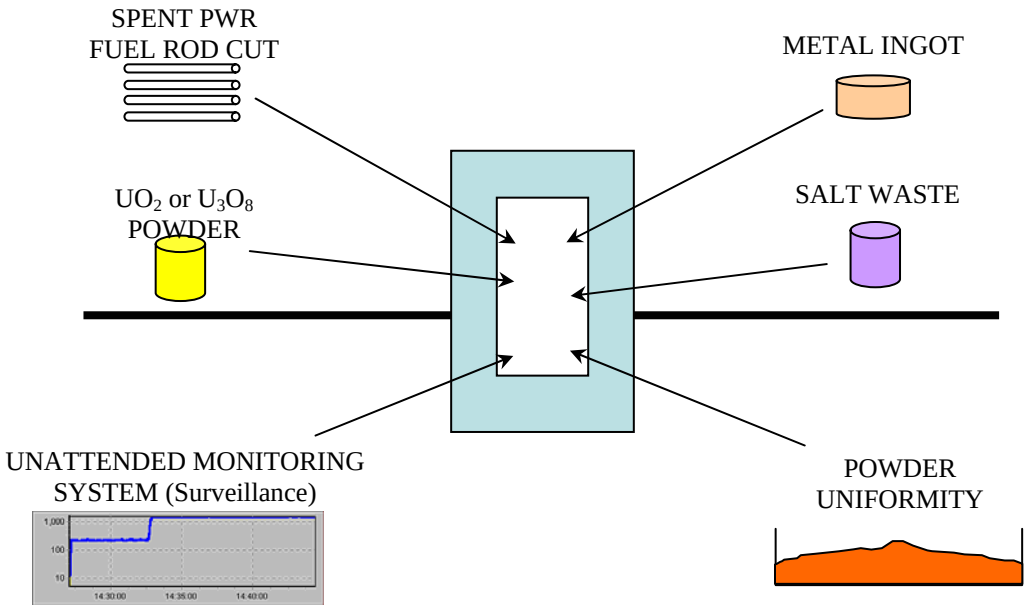

Fig. 6 Implementation and Application of ASNC

4. Remote Monitoring System

The role of remote monitoring is getting increase as a vital component of safeguards. Remote monitoring has been identified as a realistic technique to increase the effectiveness and efficiency of safeguards implementation. It is expected that it can achieve substantial reductions in inspection labor hours and exposures in hazardous environments such as high radiation. However, it also has potential drawbacks such as possible loss of information surety, and heavy initial investment and transmission cost. Considering the future of remote monitoring that is closely related to Integrated Safeguards (IS), the DUPIC safeguards team has conducted the related research activities for the non-proliferation transparency of DUPIC facility.

Recently, Virtual Private Network (VPN) has emerged as a solution to data security problems, which allowing the Internet to be a viable data transmission method for remote monitoring. For the feasibility of remote monitoring of DUPIC facility, the feasibility of VPN technology for the remote monitoring of DUPIC facility has been tested and the surveillance data could be transferred to the central monitoring station in TCNC (Technology Center of Nuclear Control) via local intranet as shown in Fig. 7. The R&D efforts for the advanced C/S system will continue to stabilize and enhance the performance of the system through the integration of gamma monitoring system and remote monitoring using VPN. The remote monitoring system will be implemented to ACP lab-scale facility in a couple of years.

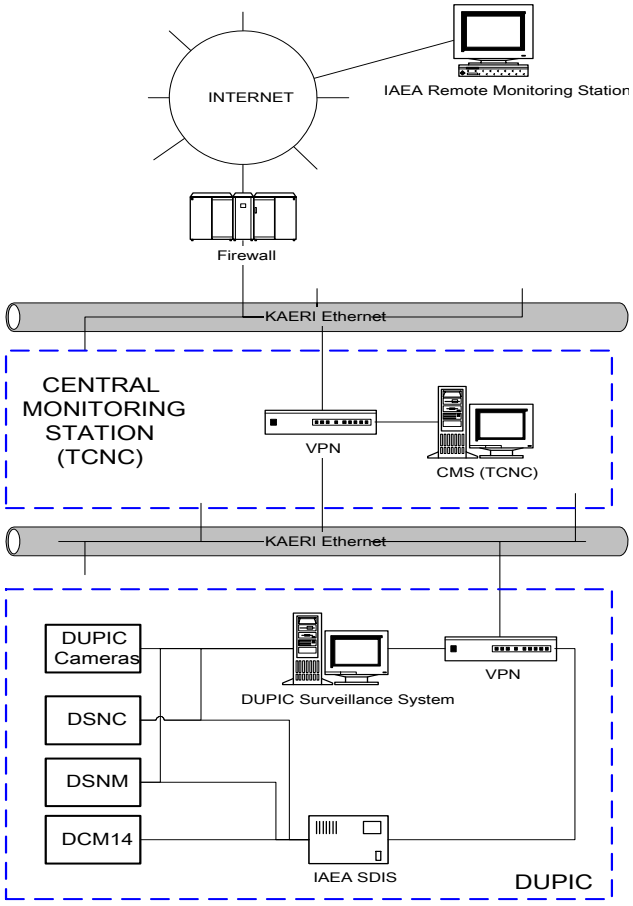

Fig. 7 Design of DUPIC Remote Monitoring System using VPN.

5. Summary

KAERI has been developing the DUPIC fuel cycle and ACP technology in the spent fuel bulk handling facility. A lab scale DUPIC facility safeguards system was successfully established under the

international cooperation program. With the implementation experience gained from the lab scale facility and further advancement of safeguards technology development, a model safeguards system for the pilot scale bulk handling facility is foreseen in the future. The R&D efforts for ACP safeguards system will continue to stabilize and enhance the performance of the system through the integration of material accounting monitoring system and remote monitoring using VPN. It is expected that remote monitoring will become a vital tool in enhanced cooperation of streamline inspection efforts of the IAEA and the Korea. It also expected that the remote monitoring system under construction in DFDF and ACPF could contribute to the reduction of inspection time and efforts, and for non-proliferation transparency of DUPIC and ACP study.

6. References

[1] Kim, H.D., et al., The Performance of the DUPIC Safeguards Neutron Counter; 43rd INMM Annual Meeting, Orlando, USA; 2002.
[2] Ha, J.H., et al., The Development of a Neutron Counter System for the Nuclear Material Accounting and Safeguards in KAERI; Journal of Nuclear Science and Technology, Supplement 4, 381-3; 2004.
[3] Kim, H.D., et al., Intelligent Surveillance System for DUPIC Fuel Development Facility; 42nd INMM Annual Meeting, Indian Wells, USA; 2001.
[4] Li, T.K., et al., Safeguardability of Advanced Spent Fuel Conditioning Process; 45th INMM Annual Meeting, Orlando, USA; 2004.

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Ho-dong Kim

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Prediction of the Consequences of the Detonation of a Radiological Dispersal Device ("Dirty Bomb")

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Joint Research Centre

Effect of a Radiological Dispersal Device (RDD, also called "Dirty Bomb")

→ disperses radioactive substances with the help of conventional explosives

- kills only a few people as a result of the initial blast
- creates panic and chaos (weapons of mass disruption/dislocation/effect)
- radioactive material is transported attached onto aerosol particles
- health damage of the population because of external radiation and inhalation
- contamination of a large area can result in severe consequences for the infrastructure (buildings are not usable for a certain time)
- consequences for the economy are significant (decontamination of a large area requires a huge amount of resources)

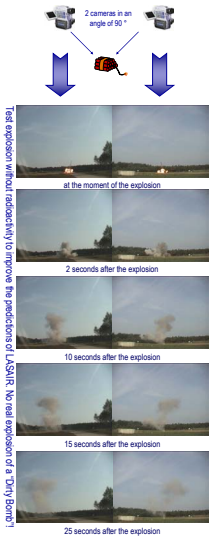
History of Dirty Bombs


The cesium-filled package uncovered in Moscow 1995

- 1995: Chechen rebels placed an RDD in the Izmailovsky Park in Moscow near the Kremlin (containing ^{137}Cs), it was disarmed
- 1998: a container of radioactive material was found attached to an explosive mine near a railway line not far from Angun in Chechnya; it was defused
- 1993-2003: 540 confirmed cases of illicit trafficking of radioactive and nuclear material, 182 of them with nuclear material[1]

[1]IAEA database, please see for more details http://www.iaea.org/NewsCenter/Features/RadSources/Fact_Figures.html

Consequences of the detonation: the formation of a radioactive cloud



Explosion of 10 kg PETN (Pentaerythritol Tetranitrate) with iron oxide powder to describe the formation of the cloud (German Federal Office for Radiation Protection [BfS])

The dispersion of the radioactive cloud

- Emergency management tools to describe large-scale phenomena (e.g. an accident in a nuclear power plant) are not suitable to properly describe the small-scale phenomena of a "Dirty Bomb"
- Local-scale model was developed for potential application in Germany by the German Federal Office for Radiation Protection (BfS): LASAIR (Programmsystem zur Lagrange-Simulation der Ausbreitung und Inhalation von Radionukliden)
- LASAIR: software simulation to predict the dispersion of radioactive substances:
  - simulates the transport of dispersed radioactive and nuclear substances attached onto aerosol particles
  - predicts the inhalation dose
  - estimates the contaminated area
- Consequences of a Dirty Bomb will be less severe, if decision makers have sufficient information and can provide coordinated countermeasures

Goal of the LASAIR collaboration

- use of the existing technology to describe the hazardous potential of a "Dirty Bomb" and application for European territory
- strengthen cooperation with international organisations (Europol, WCO, UNICRI, OSCE, SECI, Interpol, OPCW, ITWG, IAEA) and support regional cooperation
- comparison with other existing models
- improve situation awareness and support to decision-makers

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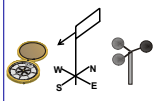
LASAIR collaboration

Prediction of the radioactive cloud after a Radiological Dispersal Event

Measurement of the dispersed nuclide on the spot with mobile equipment

Creation of a digital map of the surroundings of the spot

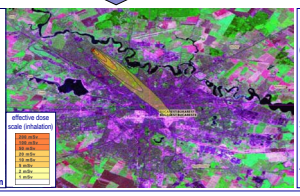
Information about wind and weather condition



Simulation of the dispersion in the environment

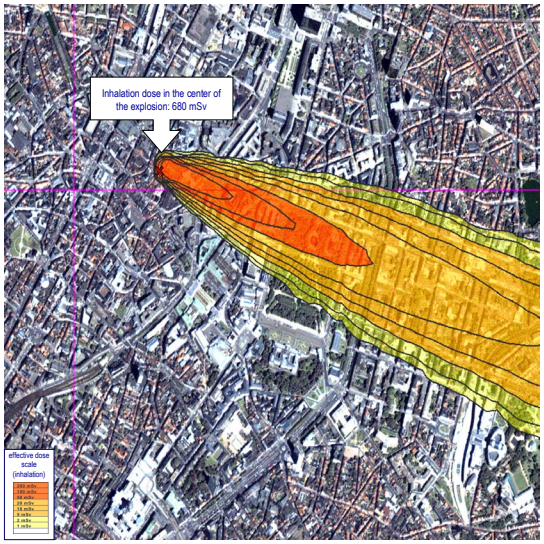
Estimation of the approximate amount of explosive by law-enforcement services

Simulation of a RDD with 4000 Ci ^{60}Co ($1.5 \cdot 10^{14}$ Bq), 15 kg conv. explosives, wind with 3 m/s from 310° north, neutral/stable weather conditions, LANDSAT digital satellite images in a scale of 1:50,000



Prediction of the dispersion of the radioactive substances in the environment to support decision-makers and minimize the consequences

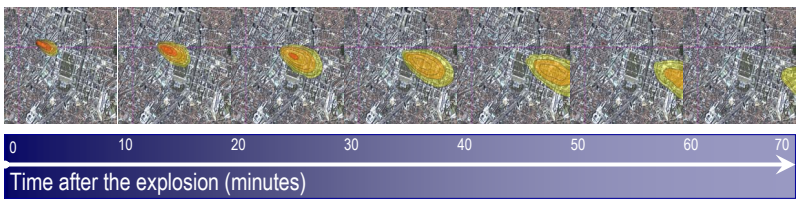
Utilization of digital satellite images to describe the dispersion in the environment



Simulation of an RDD in a major European city with 2000 Ci ^{60}Co ($7.4 \cdot 10^{13}$ Bq), 1.5 kg of conv. explosives, wind with 0.5 m/s from 290° north, weather conditions neutral/unstable; scale 1:10,000; IKONOS digital satellite image of the Digital Map Archive; shown is the total inhalation dose for 2 hours

No real explosion.

Propagation of the radioactive cloud



Simulation of an RDD in a major European city with 2000 Ci ^{60}Co ($7.4 \cdot 10^{13}$ Bq), 1.5 kg of conv. explosives, wind with 0.5 m/s from 290° north, weather conditions neutral/unstable; scale 1:10,000; IKONOS digital satellite image of the Digital Map Archive; shown is the inhalation dose for time intervals of 10 minutes each

No real explosion.

Results of the LASAIR collaboration

- Cooperation between ITU, IPSC (both JRC) and BfS results in the efficient use of the existing resources
- Experts from different areas combine their skills to apply LASAIR on European territory using digital satellite images (e.g. LANDSAT)
- Further co-operation will lead to a better description of this small-scale phenomenon

→ The LASAIR collaboration has now the ability to perform a calculation for a "Dirty Bomb" attack for, in principle, every location worldwide using satellite images.

The support of the following persons is gratefully acknowledged: S. Abousahl, V. Berthou, Omer Cromboom, Klaus Mayer, Herbert Ottmar, W. Wagner (ITU), D. Al-Khudhairy (IPSC), G. Heinrich (BfS)

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The Use of High Efficiency Neutron Detectors for Making Secondary Plutonium Working Standards

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ABSTRACT

Neutron coincidence counters are used in safeguards to measure plutonium content in many forms of process materials. Normally, standards containing known amounts of plutonium are used to calibrate the neutron detectors for the amount of Pu-240 effective in the different sample categories. When the calibration standards are produced and characterized by destructive analysis (DA), the cost is high and the material forms are limited to sample categories that can be reduced to a form for DA processing. The DA standards must quantify the plutonium mass as well as the plutonium isotopic ratios so that the Pu-240 effective can be determined for the subsequent coincidence count.

This paper shows how working standards can be created using the Epithermal Neutron Multiplicity Counter (ENMC) [1] *to directly determine the Pu-240 effective mass in typical working materials. After a careful measurement in the ENMC, the working material can be used as a secondary standard in any other NDA system. The assay in the ENMC is independent of potential impurities in the working material and the measurement provides the Pu-240 effective mass as well as the neutron multiplication (M). The reference calibration of the ENMC is not dependent of the material in the secondary standard and it can be independently verified by the Inspectorates. Bulk samples of plutonium powder or MOX can be measured by the ENMC while still inside the glove box containment and then transferred to an installed NDA system to calibrate the second system such as a holdup measurement. Waste materials such paper towels and rags can be placed in plastic bags and measured in the ENMC. Subsequently the bags can be loaded into a 200l drum to be a standard for a waste drum counter.*
The accuracy for determining the Pu-240 effective mass in the secondary standard is about 0.5% for small samples in the range from 0.1–50g and ~ 1-2% for larger samples. The counting time to establish the standard masses is in the range of 10 min to one hour depending on the mass value. The multiplicity counting in the ENMC is an order of magnitude faster than for typical neutron well counters because of the high efficiency (65%) and the short die-away time (19 micros). In addition, the ENMC provides a supplemental measurement of the average neutron energy from the sample so that impurity components that effect the energy can be flagged.

Keywords: Neutron multiplicity measurements, plutonium standards, NDA calibration, waste measurements

1. Introduction

Neutron detectors using ^{3}He tubes have been used extensively for quantitative measurements of uranium and plutonium for safeguards and nuclear material accountability.[2] The measurements are made using neutron time-correlation (coincidence) counting or multiplicity counting to obtain the plutonium or uranium mass. To improve the measurement precision and reduce the measurement time, the detectors need to have high efficiencies and short neutron die-away times. We have developed a high-efficiency neutron detector that uses 10-atm He-3 tubes in place of the conventional 4-atm tubes to count the epithermal neutrons in addition to the thermal neutrons. Experimental tests have shown that the 10-atm tubes increase the efficiency and decrease the die-away time so that the multiplicity measurement time is decreased by an order of magnitude for high alpha samples. The use of the Epithermal Neutron Multiplicity Counter (ENMC)[3] has been extended for the creation of secondary standards in MOX fabrication facilities.

2. Basis for Production of Secondary Standards

The present application of the ENMC is to establish secondary standards from in-plant working nuclear materials. For impure PuO_2 or MOX samples, the α value (ratio of (α,n)/spontaneous fission) is unknown, and it is necessary to use multiplicity counting to measure the α value and the Pu-240 effective mass. In the multiplicity mode, we measure Singles (S), Doubles (D), and Triples (T) to solve for ^{240}Pu, α, and M (multiplication). Thus, the impurity level in the sample that causes alpha to vary is measured together with the mass and multiplication.

The high efficiency and short die-away times of the ENMC are required to keep the statistical error in D and T small. Because the Singles are used in the analysis, the room background neutron S rate must be measured to subtract from the sample measurement. The D and T background rates are from cosmic-ray spallation and they are normally negligible. The statistical precision required for the creation of secondary standards is in the range of 0.2% to 0.5% so that the standards can be used for bias-defect type measurements. Precision and accuracy values of ~0.3% have been achieved as part of the ERATOM Luxembourg's On Site Lab (OSL) program[4] for pure PuO_2 and MOX inventory samples. The present work extends the NDA capability to impure materials and bulk samples.

3. Detector Description

3.1. Helium-3 Tubes

The key technical improvement resulting in the high-efficiency, short-die-away time detectors is the development of ^{3}He tubes with 10-atm pressure and fast-pulse collection time. The tubes have proprietary gas additives that are required to give a fast charge collection that is compatible with the use of AMPTEK amplifiers.[5] The ENMC uses 121 Reuter-Stokes tubes with a diameter of 25 mm and an active length of 711 mm.

3.2. Moderator and Electronics

The detector design was developed using the MCNP code[6] where the design goals were a maximum efficiency with a minimum neutron die-away time. We reduced the high-density polyethylene (HDPE) thickness between tubes to 10–15 mm and added an annulus of Cd on the inside and outside of the outer ring of He tubes to obtain a die-away time of 19.1 µs for the time region that brackets our coincidence gate of 24 µs. The sample cavity is 193 mm (7.62”) in diameter. Figure 2 shows the He-3 tubes and the polyethylene moderator, and Fig 3 shows the assembled detector.

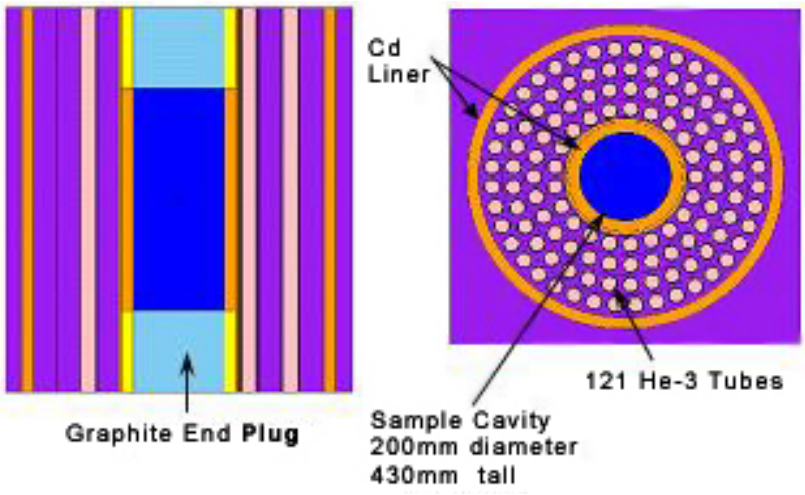

Fig. 1. Diagram of the ENMC showing the layout of the ^{3}He tubes in the HDPE detector body.

Fig.2. Photograph of the ENMC He-3 tubes partially withdrawn from the polyethylene moderator.

3.3. Multiplicity Electronics

The ENMC can operate in both the standard coincidence mode as well as the multiplicity mode. In both modes, the primary source of statistical uncertainty is the pileup of accidental counts in the coincidence gate. To reduce the dead time, there are 27 AMPTEK (A111) amplifiers to reduce the pulse rate through each amplifier. The multiplicity counting is required to make secondary standards from impure plant working materials.

3.4. Inspectorate Authentication

For storage of the ENMC between measurement use, the detector head and the electronics cabinet can be put under inspectorates seals. Figure 3 shows the ENMC with the stainless steel security cover in place. The LED lights and the desiccant tubes are visible through the side window that is under seal. Both the cabinet and detector are portable for transfer into storage room.

Fig. 3. Complete ENMC system with security cover and end plug.

4. Performance Characteristics

4.1 HV Plateau

A series of measurements were made using ^{252}Cf and plutonium to characterize the detector. The S, D, and T plateaus for the sum of the 27 amplifiers are shown in Fig 4, where the plateau operating voltage is 1720 V. The gains in all of the amplifiers have been matched so that the plateau from a single amplifier is the same as for the sum of the 27 amplifiers.

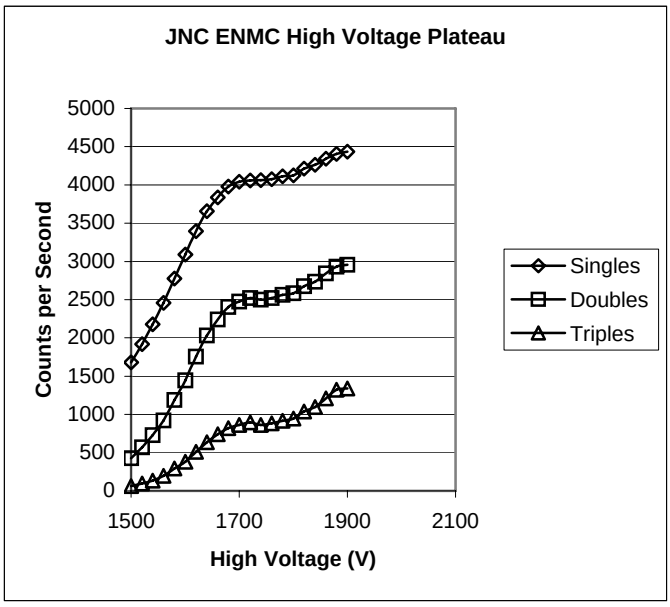

Fig. 4 . High-voltage plateau curve for S, D, and T measured in the ENMC.

TABLE I. ENMC Performance Parameters	
Parameter	Value
ENMC efficiency (^{240}Pu energy)	0.640
Die-away time (ENMC)	19.1 µs
Operating bias	1720 V
Dead-time coefficients	
A	0.121
B (A/20)	0.0061
C	0 (TBD)
Multiplicity dead time	~35.0 ns
Gate	24 µs
Predelay	1.50 µs
Doubles gate fraction	0.621
Triples date fraction	0.400
ρ_0 constant	0.346

4.2. Efficiency

We measured the efficiency of the ENMC using calibrated ^{252}Cf sources. The efficiency for ^{252}Cf was 62.5%. The efficiency was measured using small plutonium sources to be 64.0% for ^{240}Pu.

For the conventional two-parameter analysis of neutron coincidence data, it is useful to define the multiplication constant ρ_0, where

$$\rho_0 = \frac{R}{T}(1+\alpha)$$ (for a non-multiplying sample),

where *rho-naught* is the calculated ratio of alpha-particle-induced neutrons to spontaneous-fission neutrons.

Small MOX pellets and a 0.695-g enriched ^{240}Pu sample were used to measure ρ_0, giving 0.346, for a predelay of 1.5 µs and a gate length of 24 µs.

The multiplicity analysis does not use the ρ_0 constant; however, the "known alpha" method does. This known alpha method could be used for MOX samples that are pure so that alpha can be calculated from the plutonium isotopics.

4.3. Gate Length

In the past, most neutron coincidence counters used for IAEA inspection have operated with a pre-delay gate of 4.5 μs, and multiplicity counters have used 3.0 μs. However, to take full advantage of the fast die-away time for the 10-atm tubes, we have reduced the pre-delay gate for the ENMC to 1.5 μs. A series of measurements using AmLi random neutron sources were performed to evaluate if there is a bias for short predelay settings. Because the source emission from AmLi is random, the net D should be zero. The experiments showed that the ENMC bias was negligible (< 0.1%) for a predelay value of 1.5 μs or greater.

To investigate the optimum gate setting for doubles counting, we counted a small sample for a fixed counting time (30 x 20 s) using gate settings from 8 to 80 μs. The relative statistical error on the D and T rates is shown in Fig. 5. We see that the minimum error is between a 20–25 μs gate setting and that is consistent with the die-away time of 19.1 μs.

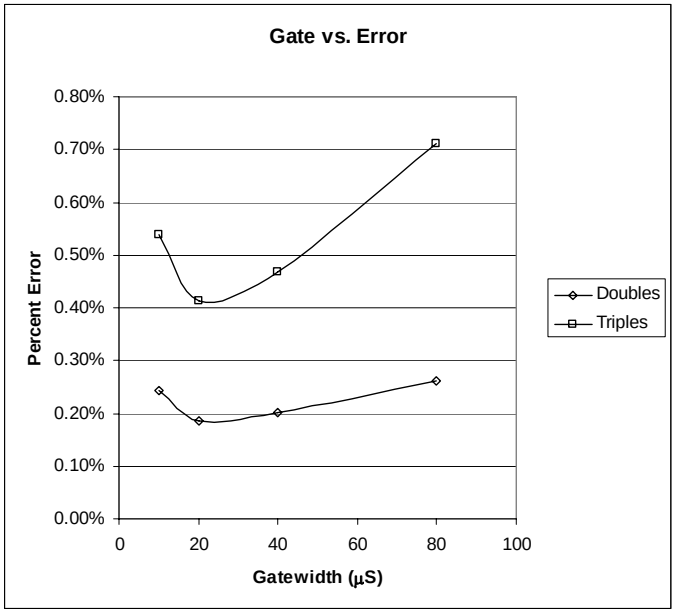

Fig. 5. Statistical error as a function of gate setting for a small Cf-252 sample.

4.4. Stability

The stability of the ENMC was tested by repetitively counting a PuF source for 63 cycles of 900 s each. The measured precision for the singles rate was 0.0145% over the 28-h period. A plot of the data is shown in Fig. 6, where each data point represents a 900 s count interval. For the counting rate of 62,060 cps, the predicted RSD from counting statistics alone is 0.0134% for the 900 s intervals. This is only slightly smaller than the measured RSD of 0.0145%, showing that the ENMC electronic stability is ~0.005%.

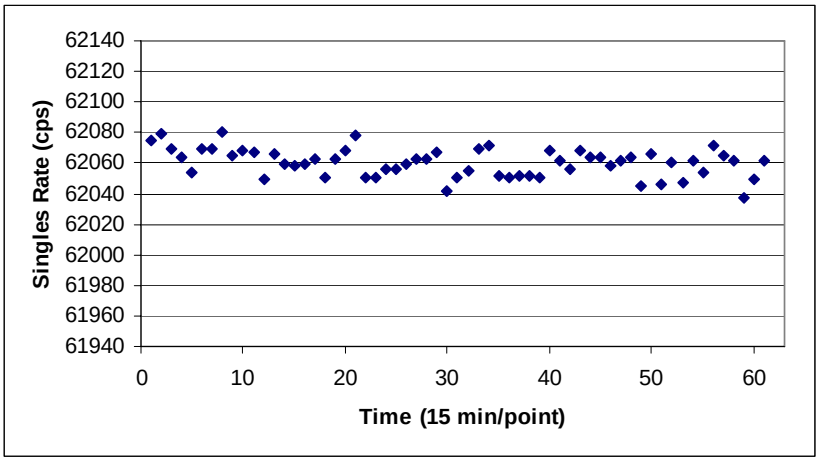

Fig. 6. Stability data for a 28-h counting interval for an AmLi neutron source.

4.5. Ring Ratios for Average Neutron Energy

The ENMC has separate signal outputs for each of the four rings of tubes. These outputs are fed into the auxiliary inputs to the AMSR. Only the S rates are available from the individual rings. The ring ratio provides an approximate average energy for a source in the cavity as illustrated in Fig. 7. This energy information provides additional information on the impure scrap samples.

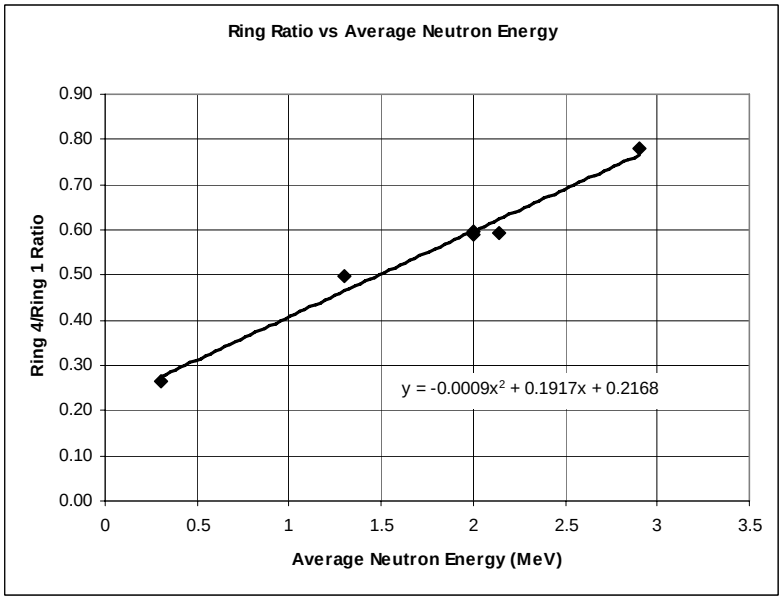

Fig. 7. Average energy calibration for the ENMC ring ratio (ring 4/ring 1).

5. CALIBRATION

Table II lists standards that were used for the ENMC calibration at LANL. These standards include separated Pu-240 to minimize any neutron multiplication, and PuO_2 combined with impurities to increase the alpha value and the induced multiplication

TABLE II. LANL standards for the ENMC calibration

Standard ID	Material	Multiplicity Alpha	Total Pu (g)	Isotopics Date	Pu-239 (g)	Pu-240eff. (g)
FZC-158	Pu-240 oxide	Low (0.169)	0.695	78/12/15	0.006	0.705
P240	Pu-240 oxide	Low (0.184)	55.57	02/12/04	3.22	52.2
646078	MOX pellet	Medium (1.08)	0.8061	91/08/30	0.7128	0.089
646081	MOX pellet	Medium (1.09)	0.5077	91/08/30	0.4490	0.056
646078+081	MOX pellet	Medium (1.08)	1.314	91/08/30	1.1620	0.145
646119	MOX pellet	Medium (0.813)	0.2651	91/08/30	0.2311	0.033
PuOC-1	PuO2	Medium (0.917)	2.002	02/05/23	1.881	0.119
PuOC-2	PuO2	Medium (0.916)	4.971	02/05/23	4.671	0.295
PuOC-3	PuO2	Medium (0.913)	9.935	02/05/23	9.337	0.589
PuF-A1	Pu metal	Low (0.040)	1.765	00/01/01	1.658	0.105
LAO250C10	PuO2	Medium (0.525)	59.84	83/09/09	49.57	9.934

The Standards listed in Table II have a variety of chemical composition and simulated impurities. However, they are all on the low end of the mass range, resulting in low multiplication. For large mass samples, there will be both neutron absorption and multiplication that will change the detector response. For the production of secondary working standards, the ENMC should be calibrated with reference samples that are in the same mass range as the unknowns. However, the calibration change in going from small to large samples is only ~ 2%.

The multiplicity calibration procedure is to measure the standards listed in Table II, and do a best fit of the results to determine the ENMC efficiency, doubles, and triples gate fractions. These parameters are then used on all future measurements until a recalibration is performed. The stability of the ENMC is good enough (~0.01%) so that recalibrations are not normally required for periods of several years. A reference neutron source can be measured on some frequency (~monthly) to confirm the stability assumption.

The ENMC was calibrated using PuO_2 standards and mixed oxide (MOX) standards listed in Table II. Table III gives the results of the calibration measurement, and Fig. 8 shows a plot of the D and T as a function of the ^{240}Pu effective mass, where the

$$^{240}\mathrm{Pu}_{\mathrm{eff}} = 2.52\ {}^{238}\mathrm{Pu} + {}^{240}\mathrm{Pu} + 1.68\ {}^{242}\mathrm{Pu}\,.$$

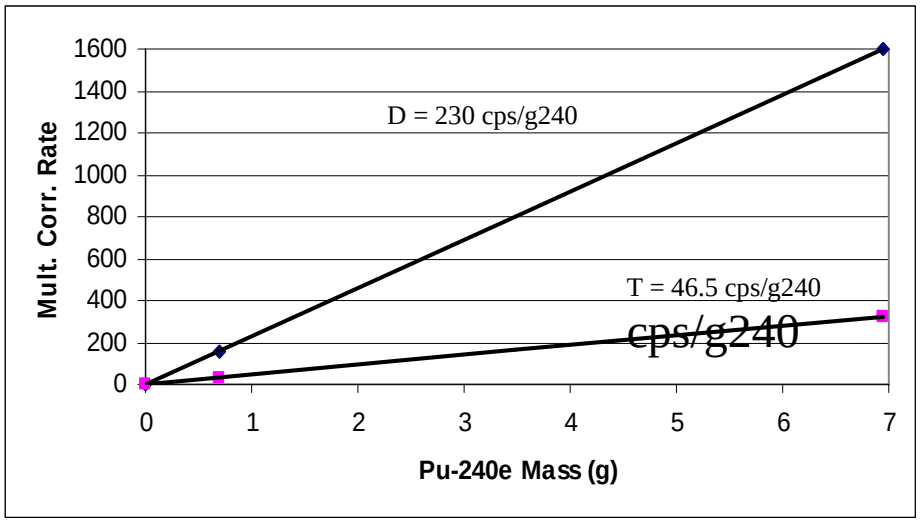

Fig. 8. Calibration response lines (D and T) for small PuO_2 and MOX standards.

TABLE III. Calibration Results for ENMC

Sample	^{240}Pu-eff (g)	Time (s)	S (counts/s)	D (counts/s)	T (counts/s)	D_{mc} (counts/s)	D_{mc}/g-$^{240}Pu_e$
FZC-158	0.705	35 x 100	530.45	160 ± 0.39	31.3 ± 0.23	160 ± 0.39	231.98
P240	52.2	10 x 30	42843.26	15947 ± 17	5347 ± 38.4	12070 ± 6.12	232.00
646078	0.089	117 x 100	120.1	20.91 ± 0.071	4.231 ± 0.049	20.91 ± 0.071	232.33
646081	0.056	210 x 100	75.45	13.02 ± 0.044	2.626 ± 0.046	13.02 ± 0.044	232.50
646078 + 646081	0.145	140 x 100	195.3	33.88 ± 0.088	6.819 ± 0.077	33.88 ± 0.088	232.08
646119	0.033	58 x 100	38.49	7.723 ± 0.060	1.596 ± 0.030	7.723 ± 0.060	234.03
PuOC-1	0.119	35 x 100	146.5	27.34 ± 0.172	5.436 ± 0.121	27.34 ± 0.172	231.69
PuOC-2	0.295	36 x 100	368.2	71.14 ± 0.259	15.21 ± 0.130	71.14 ± 0.259	231.72
PuOC-3	0.589	35 x 100	739.7	146.7 ± 0.371	32.99 ± 0.302	145.3 ± 0.154	232.15
PuF-A1	0.105	36 x 100	69.80	24.50 ± 0.156	5.297 ± 0.156	24.04 ± 0.128	231.17
LAO250C10	10.07	45 x 30	10111	2547 ± 3.237	602.4 ± 2.959	2295 ± 1.299	227.90

The multiplicity calibration is determined from a best fit of the well characterized standard samples. The efficiency was determined to be 64.0% and the doubles gate and triples gate were adjusted to provide a best fit to the standards when measured in the multiplicity mode. The calibration constants were used to analyze the samples listed in Table II to see the scatter in the

calibration fit. We have plotted the difference ratios in Fig. 9 for the multiplicity method. The Known Alpha method shows large errors for the impure samples that have incorrect alpha values based on the calculation from the isotopic vectors. The multiplicity results agreement to the known Pu-240 masses is good with a RSD of 0.8%. This data demonstrates that a large variety of samples can be accurately measured with a single set of multiplicity calibration constants.

Table IV. Assay Results			**Known α**		**Multiplicity**		
Source ID	**Singles (cps)**	**Declared ^{240}Pu (g)**	**^{240}Pu (g)**	**A-D/D %**	**^{240}Pu (g)**	**A-D/D %**	**Measured α**
646081	75.455	0.056	0.056	0.48%	0.055	-1.79%	1.091
646078	120.105	0.089	0.090	0.94%	0.088	-1.12%	1.075
PuOC-1	146.512	0.119	0.117	-1.28%	0.117	-1.68%	0.917
646078+646081	195.950	0.144	0.146	1.46%	0.144	0.00%	1.077
PuOC-2	368.202	0.295	0.307	4.07%	0.291	-1.36%	0.913
FCZ158	531.031	0.695	0.691	-0.57%	0.693	-0.29%	0.170
PuOC-3	739.691	0.583	0.627	7.52%	0.583	0.00%	0.901
87-000	2,784.264	0.562	0.618	9.96%	0.562	0.00%	6.478
LAO250C10	10,063.656	10.072	9.906	-1.65%	9.95	-1.21%	0.524
P240	42,832.000	52.313	52.066	-0.47%	51.899	-0.79%	0.183

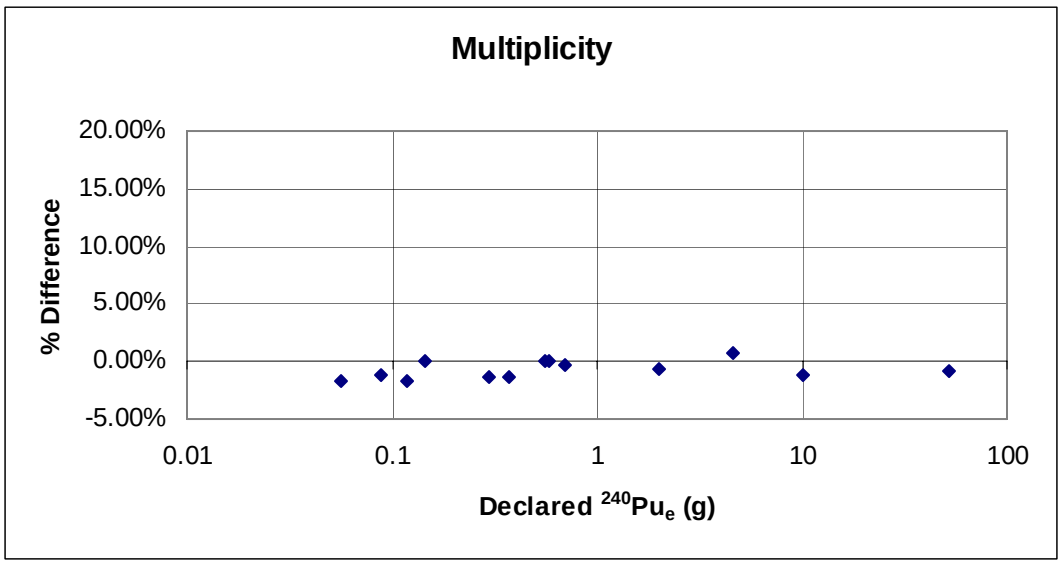

Fig. 9. The percent difference between the measured Pu-240 mass versus DA reference mass for the standards listed in Table V using the multiplicity method.

7. Applications

7.1. Secondary Standards for NDA

The ENMC has an absolute calibration that is insensitive to the sample size or composition. This opens the possibility of using the ENMC to calibrate "secondary standards" for NDA applications. The plutonium mass range for the secondary standards can range from a few milligrams to kilograms of production type materials. To create secondary standards with an accuracy that is better than 1%, it is necessary to calibrate with a standard that is in the same mass range. It is necessary to have the Pu isotopic vectors from mass spectrometry or a separate HRGS measurement.

The procedure is to measure the proposed secondary standard for a time interval that gives ~0.5% statistical errors in the mass value. This time will typically be 10–60 min. The ENMC measurement also gives the neutron multiplication M and the alpha value.

For creating secondary standards, the primary application of the ENMC is for bulk samples of plant working material. The containers can be metal or plastic, with or without plastic bagging.

Typical measurement times are 10 min. for low alpha samples to an hour for very high alpha samples. The higher mass samples typically take longer to measure than the low mass samples because of the statistical precision for the T rate.

7.2. On-Site Laboratory Use

The ENMC can be used in place of INVS type detector for those cases where the grab sample can be removed from the glove box. For on-site laboratory (OSL) applications, it is normal procedure to operate the INVS-IV[4] in the D mode without the multiplication correction because of uncertainties in the S room background. The small samples have a low S rate, and thus a few percent changes in the room background can cause a significant error in the net S. The desired measurement accuracy at the OSL is ~0.3%, and so the errors from counting statistics and multiplication must be very small. The ENMC has a factor of 5 reduction in counting time advantage over the INVS-IV. The counting time reduction is a result of the higher efficiency and short die-away time in the ENMC. The counting time to reach a given precision is illustrated in Fig. 10 for a 3gram sample of ^{240}Pu-eff.

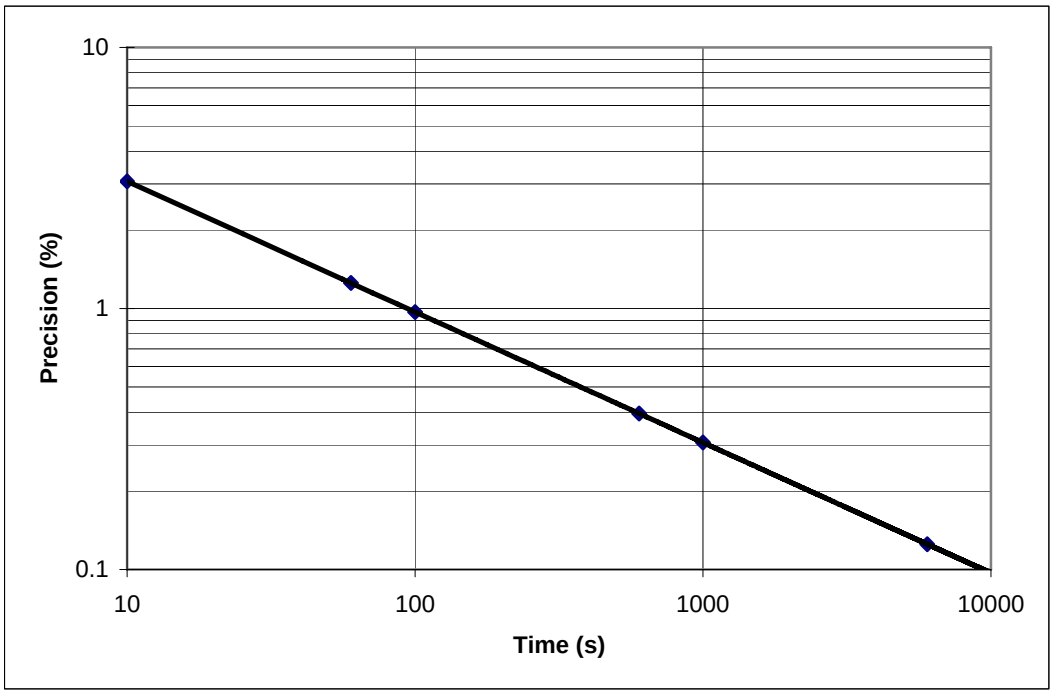

Fig. 10. Counting time versus precision for the ENMC.

The INVS was designed to fit around a well in the bottom of the glove box in the analytical laboratory, and thus the sample could be lowered into the detector cavity without bagging it out of the glove box. The ENMC is larger than the INVS, and it would require a custom box design to avoid the bag out procedure.

8. Summary

We have developed the ENMC for the measurement of impure plutonium and MOX samples. The use has been extended to small grab samples and to create secondary standards from production materials. The detector has excellent performance characteristics. The parameters that determine the performance of neutron multiplicity detectors include the efficiency, the die-away time, the count-rate capabilities, the stability, and the resistance to gamma-ray interference.

For the ENMC, the efficiency is 64%, the die-away time is 19.1 μs, the maximum counting rate is ~2.0 MHz, the stability is 0.01%, and gamma-ray resistance is ~1 R/h for the sample on contact. The most important parameter for multiplicity counting is the efficiency because the T rate varies with the efficiency cubed. If the efficiency is not extremely stable, the multiplicity counting will have large errors.

The ENMC can measure a small sample to a given precision in a time that is ~5 times faster than the INVS. For multiplicity counting of a high α sample, the timesavings is approximately a factor of 10. The precision that can be achieved for small samples with the new ENMC is similar to what can be achieved with calorimeters, and the counting times are shorter.

The ENMC has the capability to create secondary standards from production materials. The ENMC assay is relatively insensitive to the mass, shape, and chemical composition of the sample, so the calibration provided from a known standard can be used to quantify other materials. The inspection authorities could use the ENMC to certify or recertify bulk samples for use in other NDA systems in the facility. For example, bulk MOX standards of about 500 g plutonium each that are used for holdup calibration could be recertified for the IAEA use when a new glove box comes online. The ENMC measurement also provides the neutron multiplication that is needed to correct for the bulk sample multiplication compared with the holdup that has negligible multiplication.

IX. Acknowledgements

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References

1. J. E. Stewart, et al., "The Epithermal Neutron Multiplicity Counter (ENMC): Faster Plutonium Assays by Factors of 5-20," *Nucl. Mater. Manage*. **XXVII**, July 25-29, 1999. Los Alamos National Laboratory document LA-UR-99-3791.

2. H. O. Menlove, "The Role of Neutrons in Safeguards," *JNMM* **XV (4)** 83-87 (July 1987). Los Alamos National Laboratory document LA-UR-87-1401.

3. H. O. Menlove, et. al., " Manual for the Epithermal Neutron Multiplicity Detector (ENMC) for measurement of impure MOX and plutonium samples", Los Alamos National Laboratory report LA-17088-Rev.1 (May 2004)

4. H. O. Menlove, D. Davidson, J. Verplancke, P. Vermeulen, H. G. Wagner, R. Wellum, B. Brandelise, and K. Mayer, "Design and Performance of a New High Accuracy Combined Small Sample Neutron/Gamma Detector," *Nucl. Mater. Manage*. **XII** (Proc. Issue) 872-879 (1993). Los Alamos National Laboratory document LA-UR-93-2670.

5. H. O. Menlove and J. E. Swansen, "A High-Performance Neutron Time-Correlation Counter," *Nucl. Technical*. **71**, 497-505 (November 1985). Los Alamos National Laboratory document LA-UR-84-3544.

5. J. F. Briesmeister, "MCNP-A General Monte Carlo Code for Neutron and Photon Transport, Version 3A," Los Alamos National Laboratory report LA-7396-M (September 1986).

BADMINTON

A database to store and review non-destructive assay results

CONTEXT

- Measurement performed by IRSN in the framework of French domestic safeguards.
- Quality assurance process.

MAIN FEATURES

- Measurement recording and storage without typing: the application software reads the data directly in result files using a specific grammar
- Easy data consulting.
- Measurement conditions and technical means recording in order to:
  - → ensure the traceability of measurements,
  - → allow post-inspection analysis.
- New device recording with database update.
- Item management by facility.
- User management.

SPECIFICATIONS

- Java software (*260 class*).
- Oracle database *(May 2005: 90 tables, 24 devices registered, 2500 measurements, 8000 files stored)*.
- Xml parser for reading input data.

On site measurements with transportable device

MEASUREMENTS RECORDING IN BADMINTON


Pu isotopic composition.


U enrichment.


Gross weight.


Passive neutron coincidence counting for waste measurement.


γ Spectrometry for caracterisation of plutonium waste drum.

New measurements recording

Laboratory private network

DB

Server

BADMINTON

Request

UPGRADABLE SOFTWARE APPLICATION

- Unlimited number of devices.
- All kinds of measurement systems accepted.
- Item management by facility.
- User management.

New assay recording

Purchase a new assay: update of BADMINTON

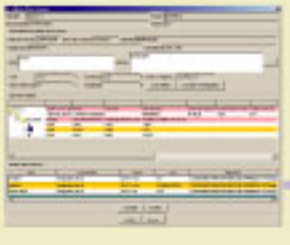
DATA CONSULTING

- Array results.
- Measurement editing.
- Whole measurement files available.

Array Export

Res.xls

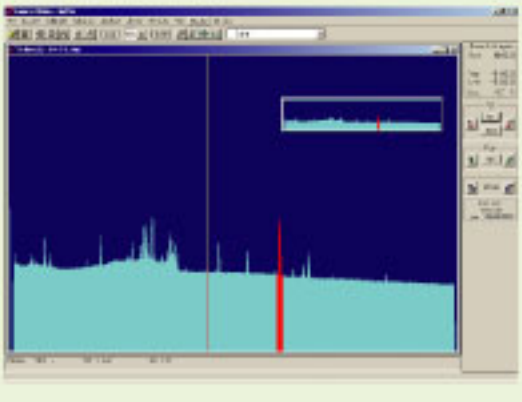
Measurement editing



Files Export

POST-INSPECTION ANALYSIS

- Spectra, text files, ... analysis with dedicated softwares.
- Re-measurement possible from measurement conditions stored in database.
- Experience feedback.
- Follow-up of laboratory measurements.



IRSN

Assay of mixed thorium/uranium contaminated waste using low-resolution gamma spectrometry

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Abstract:

During the early 1990's the UKAEA was contracted to process unirradiated THTR fuel containing a 10:1 ratio of thorium and highly enriched uranium (HEU). The work was carried out at Dounreay, and the HEU recovered from the fuel was returned to the customer. Some thorium nitrate liquor originating from this work remains stored at Dounreay.

The waste generated during this campaign is currently stored in three hundred 200-litre drums and is contaminated with a mixture of thorium and HEU. In order to sentence the waste in accordance with regulatory requirements and modern waste management standards it will be necessary to re-pack the individual waste items and obtain an accurate assessment of their radioactive content.

A low –resolution gamma spectrometry system (BIL Drumscan®) has been used at Dounreay exclusively for the assay of uranium-235 in small waste packages for the last three years. The UKAEA Non-Destructive Assay Section has recently re-commissioned this system for the simultaneous assay of uranium-235 and thorium-232. It is intended to begin re-pack operations during May 2005, at which point, individual waste packages will be assayed using this system.

This paper describes the work to re-commission the spectrometry system, including the initial technical feasibility study, calibration and testing. The typical uncertainties associated with the measurement of thorium and HEU are also given.

Keywords: Thorium, HEU, THTR, re-commission

1. Introduction

During the early 1990's UKAEA was contracted to process THTR fuel containing a 10:1 ratio of thorium and highly enriched uranium (HEU). The fuel had been intended to power a 'pebble-bed' style reactor, but could not be used since the operator did not receive regulatory consent to operate the plant. Therefore, the fuel was shipped to Dounreay in order to reclaim the HEU and return it to the customer. Much of the thorium continues to be stored in the fuel processing facility at Dounreay, in nitrate form.

There are also roughly three hundred 200-litre drums stored at Dounreay containing waste generated during the fuel processing campaign. Although the waste was originally assayed at the time of processing there is significant doubt that the figures derived for thorium-232 (the chief isotope of thorium) are accurate. Individual waste packages were assayed at the time using a low-resolution gamma spectrometry system to determine the uranium-235 content. The thorium-232 content was inferred from the 10:1 ratio of thorium-232 to HEU within the original fuel. It was inappropriate to apply this ratio uniformly to all waste packages since the exact ratio would change depending on the particular stage of processing at which the waste was generated. Furthermore, at the time of the processing campaign the assay equipment was not well characterised and the quality assurance of the equipment is questionable.

In order for UKAEA to sentence the THTR waste in an appropriate manner, meeting modern standards for waste management, an accurate assessment of its radioactive content is needed. Measurement of uranium-235 in waste packages is routinely accomplished by low resolution gamma spectrometry at Dounreay, using a system which meets quality assurance standards specified in the National Physical Laboratory (NPL) Good Practice Guide for Non Destructive Analysis [1]. However, until recently there was not an acceptable method for determining the thorium-232 content of waste. The Non Destructive Assay Section at Dounreay has re-commissioned the uranium-235 assay system for joint measurement of thorium-232 and uranium-235, to the same quality standards. Waste packages will be simultaneously assayed for uranium-235 and thorium-232 using this system, enabling the plant to achieve high throughput rates. Re-packing work is anticipated to start during May 2005.

2. Background history

2.1. Description of fuel and processing techniques

It is known that the fuel was manufactured between 1976 and 1979. Although there is little information on the production of the fuel, it is reasonable to assume both thorium and uranium would have been chemically purified at this time and hence separated from their decay products.

The fuel was manufactured as kernels of mixed thorium/uranium oxide contained within balls of graphite. The kernels were several millimetres in diameter, and encased within silicon carbide shells. Several of these kernels were contained within each ball of high-density graphite, about 10 cm in diameter.

Around 30,000 balls were processed during the course of the contract, within a specially built facility. After they had been crushed and incinerated to remover residual graphite, the fuel was dissolved in a solution of nitric acid, aluminium nitrate and ammonium fluoride. The uranium fraction of the resulting fuel solution was reclaimed by solvent extraction. The thorium fraction, with all its decay products and those from uranium too, was left in solution, much of which is currently stored in tanks within the plant.

2.2. Description of waste

The waste in the 200 litre drums was produced throughout all the stages of processing: from the initial crushing of the spheres, through dissolution and solvent extraction, to the transfer of raffinate into the storage tanks. It therefore includes all ratios of thorium-232 to HEU from 10:1 upwards.

The waste is considered to comprise swabs, glovebox filters, candle filters (used for liquids), polythene and PVC sheeting. In other words it is largely soft matrix and will not have an unduly large effect on the attenuation of gamma radiation.

The thorium contained in the waste drums has the same decay history as the material in the pot storage tanks. However as mentioned previously, the material in the drums will have HEU associated with it, whereas there is no uranium in the pot tank liquor.

3. Waste measurement strategy

3.1. Gamma spectrometry

Thorium-232 has a very low probability of spontaneous fission, and is therefore not suitable for measurement using a neutron-based assay technique. It decays via alpha emission with a half-life of 1.405 E+10 years, and since the probabilities of decay and gamma emission are very low, it is unsuitable for direct measurement via gamma spectrometry. We can, however, measure thorium-232 indirectly through its much shorter-lived decay products using gamma spectrometry, provided we understand the exact stage of equilibrium between thorium-232 and its decay products.

In order to select a suitable gamma peak from one of thorium-232's daughters, we needed first to understand what a 'typical' spectrum for the waste would look like. One of the waste drums to be repacked was assayed using high-resolution gamma spectrometry. All of the spectra were dominated by the 185.7 keV peak from uranium-235 although several peaks attributed to daughters of thorium-232 and uranium isotopes could also be seen between 890 and 1001 keV. There was also a distinct peak at 2.6 MeV, arising from thallium-208, one of thorium-232's daughters.

The choice of low resolution over high resolution gamma spectrometry was guided by two facts: Firstly, we did not have a high resolution gamma spectrometry system capable of high throughput analysis which could operate alongside the re-packing process. Secondly, there was already a low resolution system (the BIL Solutions Drumscan) installed next to where the re-packing work is due to take place already calibrated for routine assay of uranium only waste packages from different waste streams.

The 2.6 MeV peak from thallium-208 is the preferred choice for indirect measurement of thorium-232. There are no gamma rays within at least 1,000 keV of the 2.6 MeV peak, making it easy to identify and measure. Because of the convoluted nature of the spectrum at lower energies (890 to 1001 keV) low resolution gamma spectrometry is not sensitive enough to identify individual peaks arising from actinium-228 or thallium-208, for example, in this region. The 2.6 MeV peak does not sit on top of an existing Compton continuum, and therefore its ratio of net to gross counts is relatively high.

3.2. Thorium decay series and stage of equilibrium with thallium-208

Thorium-232 decays through a series of radionuclides to the stable nuclide lead-208 (see the diagram below). There are eleven members of this decay chain including thorium-232 itself, and a branch at bismuth-212 (which decays by either alpha or beta emission, with 36 and 64% probability respectively).

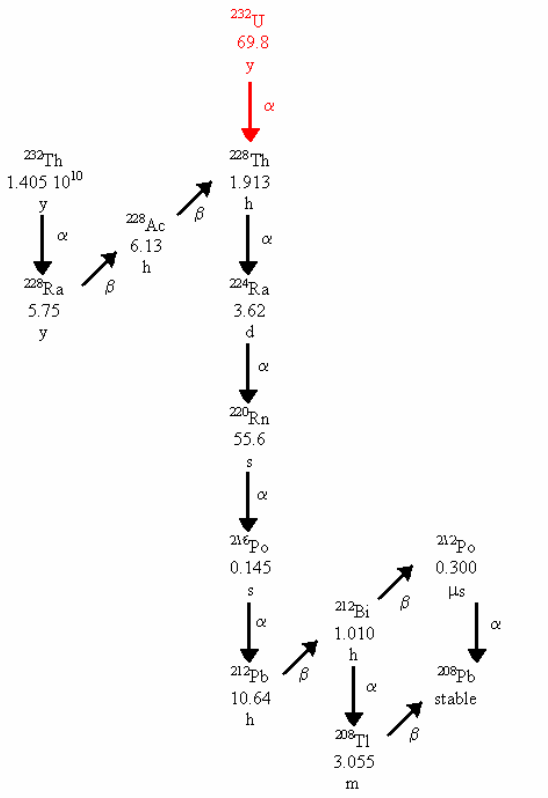

Figure 1: The decay chain of thorium-232 [3]

The stage of equilibrium between thorium-232 and thallium-208 was determined by using the Bateman equations [4] in which the activity of the nth member of a decay chain is expressed in terms of the decay constants of all the preceding members.

Figure 2 illustrates the variation of activity as a function of time for the isotopes thorium-232 and thallium-208 for an initial arbitrary sample of 100g of Th-232 and no daughter products. It can be seen that secular equilibrium (where the activity ratio, $A_{Daughter}/ A_{Th\text{-}232}$ approaches unity) is achieved after approximately 35 years. (For thallium-208 the true ratio tends to ~0.36 as a result of the branching in the chain at the preceding step.)

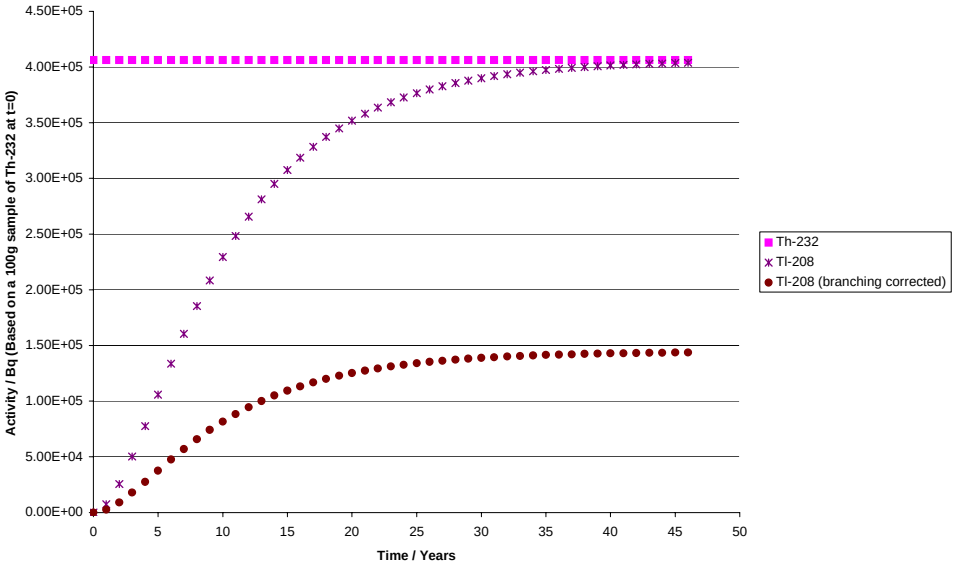

Figure 2: Variation of thorium-232 and thallium-208 activity as a function of time. The curves assume an initial sample of Th-232 of 100g with no daughter products present.

The THTR fuel, was initially constructed sometime between 1976 and 1979 as stated previously. Using the results presented in figure 2 it can be deduced that in a sample of thorium-232 of this age, equilibrium will have only been established with thallium-208 to within 9 to 6% respectively. The variation from equilibrium was accounted for by calibrating the assay system using a radioactive standard prepared from thorium nitrate solution with the same decay history as the solid waste.

3.3. Disruption to equilibrium caused by loss of radon-220 from the waste packages

Radon-220 is the fifth daughter product in the decay chain of thorium-232. Since radon is a gas, possible loss of radon from each waste package, and the subsequent effect on equilibrium between thorium-232 and thallium-208, required to be understood. In particular, the time to re-establish equilibrium between thorium-232 and thallium-208 should radon-220 gas have escape, was calculated.

To calculate an exact time for equilibrium to be re-established between thorium-232 and thallium-208 analytically is quite complex. The initial conditions required to solve the partial differential equation are quite different from those used by Bateman to form the standard Bateman equations. The partial differential equation in question is given below:

$$\frac{\partial N_{208Tl}}{\partial t} = -\lambda_{208Tl} N_{208Tl} + \lambda_{224Ra} N_{224Ra} - \frac{\partial N_{220Rn}}{\partial t} - \frac{\partial N_{216Po}}{\partial t} - \frac{\partial N_{212Pb}}{\partial t} - \frac{\partial N_{212Bi}}{\partial t}$$

where the symbols have their usual meanings. The initial rate of decay of each individual species (^{220}Rn, ^{216}Po, ^{212}Pb and ^{212}Bi) and the initial activity of radon-220's parent radium-224, would need to be known at the exact moment that radon-220 was removed from a waste package in order to solve this equation to evaluate the time required to reach equilibrium. This information would obviously not be known. Even if this information were known, the solution to the equation would be non trivial,

requiring use of complex Laplace transforms in a method similar to that used to attain the original standard Bateman equations described in the previous sections. This solution is beyond the scope of the present work and instead a more intuitive approach to the problem was employed.

A situation was considered where all the radon-220 present in a sample originally containing 100g of thorium-232 was removed at a time where equilibrium between Th-232 and Tl-208 was established to within 10%. As a result of the daughters of radon-220 having relatively short half-lives, it was decided that their contribution from the moment of removal would be zero. Instead only the new in-growth of daughters would be considered. This is obviously pessimistic but it provides us with a result of when equilibrium would be re-established. This argument is augmented further when one considers that a more realistic scenario would be the escape of a fraction of the radium-220 gas from the sample. Our theoretical situation will also over-estimate the time to re-establish equilibrium for this situation. As we are considering a situation where the radon-220 is removed at a time when the decays are within 10% of equilibrium, radium-224, the parent of radon-220, will essentially be undergoing a net decay governed by the half life of thorium-232 (as will all the intermediate species). The Bateman equations [4] were applied to this simplified system, and the results are displayed in figure 3.

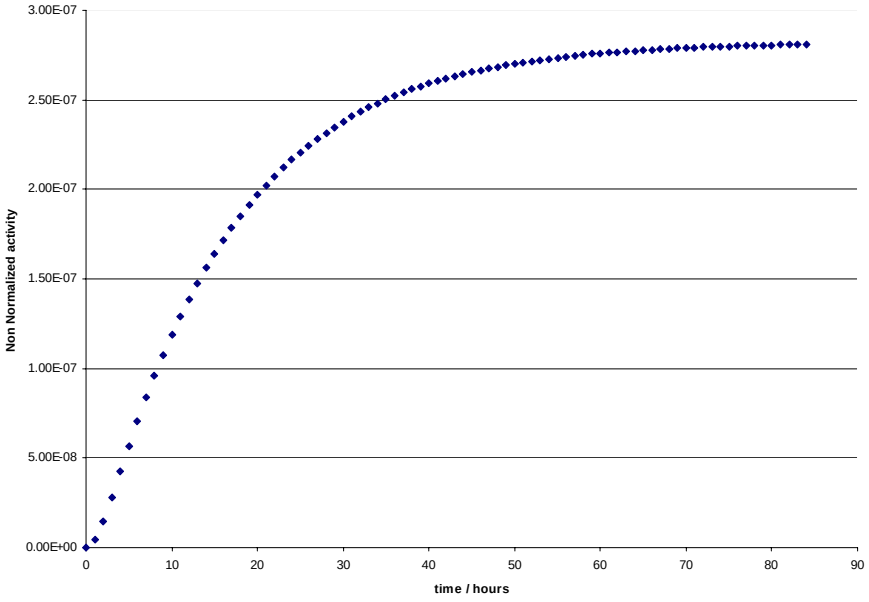

Figure 3: Variation of Tl-208 activity as a function of time after removal of Rn-220 from the decay chain. For the assumptions made in the calculation refer to the text.

It can be seen from figure 3 that equilibrium is re-established after ~50 hours (2 days) for this pessimistic situation. As we have artificially forced the half life of radium-224 (radon-220's parent) in the calculation, the activity will not tend to the activity of radium-224 as one may initially expect. Instead the activity tends towards a value determined by the ratio of radium-224 $\tau_{1/2}$ / thorium-232 $\tau_{1/2}$ (~3E-7 Bq).

It is known that all the waste packages are sealed in PVC, and during re-packing work it is unlikely that this will be opened up. However if the PVC shielding is found to be damaged, or if it is necessary to re-seal waste, then according to our pessimistic calculations on the re-establishment of equilibrium, material would only need to be left for 2 to 3 days, for full secular equilibrium to be restored.

3.4. Interference from uranium-232

Uranium-232 is formed, along with several other uranium isotopes, from the irradiation of uranium and other transuranic elements [2]. The decay scheme shown in figure 1 shows clearly that uranium-232 has decay products common to thorium-232. Therefore, measurement of thorium-232 via gamma spectrometry of any of its daughter products from thorium-228 onwards, will result in an overestimation if the material contains or is contaminated with irradiated uranium. The specific activity of uranium-232 is over one hundred million times greater than that of thorium-232, meaning that small traces of

uranium-232 in the waste could dominate the production of thallium-208 by radioactive decay and render this analysis unfeasible. Because the waste associated with the THTR fuel is known to contain HEU, the likelihood of uranium-232 being present in the waste had to be established, and quantify its effect.

Alpha spectrometry analysis records for the uranium recovered from the THTR fuel show that the uranium-232 content in each billet was less than 1 part per billion. The argument below demonstrates that even when pessimistic assumptions are applied to the effect of uranium-232 in the waste its effect may safely be ignored.

For a uranium-232 content of 1ppb and assuming a ratio of 10:1 thorium-232 to uranium, which represents the maximum uranium concentration in the waste, we calculate that for every gram of thorium-232 there were 10^{-10} grams of uranium-232. We assume that the stage of equilibrium between thorium-232 and thallium-208 is 0.97 as calculated previously, whereas for uranium-232 equilibrium with thallium-208 is near unity. In this example, 1 gram of thorium-232 has grown-in 1414 becquerels of thallium-208 whereas 0.1 nanograms of uranium-232 will have grown-in 29.8 becquerels of thallium-208. Even this pessimistic calculation shows that there is insufficient uranium-232 within the THTR fuel itself to bias the results by more than 2%.

During the THTR re-processing campaign there were no other fuels being re-processed in D1203, therefore it is extremely unlikely that THTR waste was mixed with waste derived from other fuels. Furthermore, the possibility of cross-contamination has been ruled out because of good housekeeping practices. The uranium in the THTR fuel was highly enriched (greater than 90%) and it was common practice at the time of this campaign to thoroughly clean all the processing equipment prior to the start of a HEU processing campaign. This was done to avoid contaminating the uranium with a lower enrichment of fuel and thereby decreasing its value.

4. Commissioning

4.1. The preparation of thorium standards

Samples of the thorium nitrate liquor stored in the plant were analysed by ethylenediaminetetraacetic acid (EDTA) titration to determine their thorium concentration. To prepare standards suitable for calibrating and testing the assay equipment known aliquots of the liquor were added to polythene bottles containing vermiculite, which were then double bagged in PVC. Because each of the waste packages would have to be small enough to be posted out of a conventional glovebox, the size of the standards was kept to a maximum of 1 litre. Since the waste is known to mainly comprise soft items such as tissues, polythene etc which are surface contaminated with thorium nitrate liquor these standards were considered a close approximation to the real waste. The standards were left to grow-in decay products (principally radon-220 and its daughters) for a few days prior to use.

4.2. Optimisation of detectors

Drumscan detectors are fitted as standard with a weak americium-241 gain stabilisation source. Filters around the source are used to adjust the position of the gain stabilisation peak in the gamma spectrum and typically this appears at high energy. In order to avoid a clash between the gain stabilisation peak and the 2.6 MeV thallium-208 peak, the filters around the americium-241 sources were adjusted such that the gain stabilisation peak falls at around 1500 keV.

4.3. Drumscan calibration

The calibration of the Drumscan for thorium contaminated waste followed a similar procedure for that used to calibrate it for uranium-235 measurement in uranium only packages. This procedure is described here in detail.

The Drumscan was calibrated up to a maximum mass of 35 grams thorium-232. There is a linear relationship between the recorded count-rates for all masses up to 50 grams. All calibration measurements were carried out using the 35 grams standard. Calibration measurements were

performed with the standard at various positions within the package bin, in order that we could derive volume weighted average calibration parameters for different fill heights. By doing this, we minimised the uncertainty due to position of the waste material relative to the detectors.

A MCNP™ model of the Drumscan, as supplied by BIL Solutions, was modified to include a uniform distribution of 2.6 MeV gamma emitters within a homogenous mass of varying fill height and density inside the package bin. The F8 (pulse height) tallies for these models were used to derive attenuation correction factors specific to the weight and fill height of the waste.

A series of background measurements were carried out in the plant, under typical operating conditions, and the variation in results was noted. The limit of detection was calculated from the variation in background results, which guided us towards the optimum conditions for operating the system. Management control ensures that a background count of 15 minutes is performed immediately prior to each waste package count, so that variations in the background radiation caused by re-packing operations are accounted for. Item counts also take 15 minutes to complete. The limit of detection for a typical waste package is 0.54 grams (uranium-235) and 5.4 grams (thorium-232).

Because the operators in the plant were already used to using the Drumscan for routine uranium-235 assay, they were receptive to the changes required for thorium-232 assay. In practice, the operators are only required to enter two simple data into the computer every time they carry out an assay, and the system calculates the uranium-235 and thorium-232 result based on the most appropriate calibration parameters. A uranium fingerprint associated with the waste is also applied, to infer the activity of other uranium isotopes.

4.4. Uncertainties associated with THTR waste assay

A comprehensive study into the sources of uncertainty associated with uranium-235 and thorium-232 assay was performed in accordance with the principles laid out in the NPL Good Practice Guide for Non Destructive assay [1]. Individual uncertainty components were identified and their effect on the total mesurement result was quantified as 1 sigma percentage uncertainty. The quadrature sum of these uncertainties was then doubled to give the total measurement uncertainty (TMU) with 95% confidence. The uranium-235 or thorium-232 inventory of each waste package has a marked effect on the TMU and so different combinations of uncertainty components were combined as appropriate so that several TMU values were calculated.

The uncertainties are presented in tables 1 and 2 below.

Individual item ^{232}Th mass (g)	Matrix type / mass Typical soft waste items
5.4 (= L_D)	60 %
10	35 %
35	17 %

Table 1: Percentage TMU figures for individual item thorium-232 assay expressed with 95 % confidence

Individual item ^{235}U mass (g)	Matrix type / mass Typical soft waste items
0.54(= L_D)	98 %
1	56 %
5	22 %

Table 2: Percentage TMU figures for individual item uranium-235 assay expressed with 95 % confidence

5. Conclusions

The technical assessment of thorium-232 assay using low-resolution gamma spectrometry is described in detail. Because thorium-232 cannot be measured directly using conventional gamma spectrometry techniques it was necessary to measure indirectly via one of its daughter products. Having established the stage of equilibrium between thorium-232 and one of its daughters (thallium-208) potential sources of risk to the assay strategy were identified, investigated and quantified. We are confident that thorium-232 will not be falsely overestimated in the waste packages because the analytical results for the HEU indicate a low (perhaps non-existent) uranium-232 content. Also, if during re-packing operations it is necessary to remove waste containment, the current stage of equilibrium between thorium-232, radon-220 and its daughter products may be re-established within a few days of re-packaging the waste.

The BIL Solutions Drumscan was calibrated for thorium assay using material with the same decay history as the drummed waste. There is a high level of confidence in the calibration factors derived through measurement of these standards and application of MCNP™ code. Uncertainties associated with measurement results have been calculated as well as limits of detection under typical operational conditions.

6. Acknowledgements

The authors would like to acknowledge the assistance of UKAEA staff involved at the time of the THTR processing campaign, in helping to describe the history of the fuel and the processing techniques used. The assistance of plant operators during the commissioning of the BIL Solutions Drumscan was invaluable and must also be acknowledged. Finally, our thanks are extended to the technical staff at BIL Solutions during the modification of the system software.

7. References

[1] National Physical Laboratory, *Measurement Good Practice Guide No.34 Radiometric Non-Destructive Assay*, 1, National Physical Laboratory, Teddington, Middlesex, January 2002

[2] A P Vinogradov, *Analytical Chemistry Of The Elements – Uranium*, Ann Arbor-Humphrey Science Publishers, 1970

[3] G Longworth et al, *The radiochemical manual,* 1, AEA Technology plc, Harwell, Oxfordshire, 1998

[4] H. Bateman, "*The solution of a system of differential equations occurring in the theory of radioactive transformations*", Proc. Cambridge Phil. Soc., 16, p. 423 (1910).

The Next Generation Tomographic Gamma Scanner

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Abstract

The goal of Non-Destructive Assay (NDA) of radioactive waste is to accurately identify and quantify the radionuclides present in the waste stream. Gamma ray scanning instruments such as the Segmented Gamma Scanner (SGS) have found widespread use in the NDA community. The Tomographic Gamma Scanner (TGS) aims to improve the accuracy of assay results for difficult cases such as a non-uniform distribution of radioactivity in a heterogeneous matrix. The TGS combines high resolution gamma ray spectrometry with 3-dimensional single photon attenuation coefficient images (Transmission image) and single photon Emission images.

Canberra Industries developed its prototype TGS in collaboration with Los Alamos National Laboratory (LANL). The prototype Canberra TGS was used as a Research and Development tool to conduct an in-depth study of the TGS technique. The results of these studies have been reported elsewhere.

Recently, Canberra has built its next generation of TGS systems implementing many advanced features. The new features include an automated variable collimator aperture, an automated detector slide mechanism for adjusting the sample to detector distance, automated attenuator mechanism, and an automatic selection of assay geometry based on the dose rate and density of the assayed sample. The collimator aperture is formed by interleaved layers of tungsten that can be opened and closed to change either the spatial resolution of the TGS measurement or to switch over to an SGS measurement mode. The automated slide mechanism allows the detector to move further back from the high activity drums extending the dynamic range of the system. Alternately, for use with small samples such as cans, the can can be placed on sample pedestal to allow the detector to move closer to the container for improved spatial resolution in either the SGS or TGS modes. The automated attenuator assembly is intended to allow system operation for drums with high surface exposure rates. The assembly is mounted to the front face of the detector shield and collimating aperture.

The paper discusses the features and performance of the next generation TGS.

1. Introduction

The Non Destructive Assay (NDA) of special nuclear materials in waste containers for inventory and safeguards purposes is challenging because of the potentially high degree of variability from item to item. This variability may allow the deliberate addition or removal of special nuclear materials to be made to waste containers to go undetected. These materials could be used in a clandestine nuclear weapons program. The Tomographic Gamma Scanning (TGS) technique combines transmission and emission reconstructive tomography techniques in an effort to improve the accuracy of high resolution gamma-ray spectroscopic measurements. In particular the goal is to reduce the uncertainties associated with the unknown material and density distribution of the waste matrix and also with the unknown distribution of SNM and other radioactive materials present. HRGS is retained in order to unravel the complex spectrum and in order to provide 'good geometry' results - that is peak areas which are relatively insensitive to coherent and small angle scattering

In addition to improving the quality of the assay results the intermediate transmission (linear attenuation map as a function of energy) and emission (by nuclide or line) images, although only of low spatial resolution, can provide powerful additional visual verification and confirmation information about the items.

Reconstructive tomography involves scanning the item in a series of layers or slices from many different orientations or views and using the information to reconstruct a picture of the interior of the object free from the interference effects from underlying and overlying planes. The idea is that if we know the attenuating properties of the interior along with the distribution of activity that a better matrix attenuation correction factor can be derived. The objective is to improve the accuracy of the assay while still retaining a useful detection limit and throughput at reasonable cost. Thus for a 200 liter drum, say, a representation of the item comprising 16 layers each of 88 volume elements (or voxels) is used. This contrasts with the familiar segmented gamma scanner approach which may use 8 to 16 segments, say, with each individual segment being treated as distinct but as homogeneous in matrix and uniform in activity (on a per unit mass of matrix basis). The TGS scan sequence generates a series of data grabs or views which over determine the contents of the voxel grids. Algebraic reconstruction in real space is used to extract a best fit solution consistent with the data and as free from spurious features as possible. The quality of the reconstructed images may be judged by the correspondence between the measured and actual distributions of test cases. The reconstruction model takes into account the changing collimation, attenuation and inverse square law affects at each view. These physical processes acting together provide the modulation and contrast in pattern of view data that allow the images, free from cross talk, to be formed.

The experimental realization of the TGS discussed here has been outlined elsewhere[1-3]. Here we wish to concentrate on the next generation of TGS that has incorporated in it, many new features. These improvements make the instrument suitable for assaying waste streams with a wider range of matrices and activities. We include some preliminary results generated from the new system.

2. The Next Generation TGS

Canberra industries has designed and built the next generation of TGS with the objective of making the instrument applicable to waste streams with a wider range of activities and matrices and as well as with a wider range of nuclides that is not restricted to just plutonium isotopes. Figure 1 shows a picture of the system. The TGS system can be operated in the Segmented Gamma Scanner (SGS) mode, if desired. The selection of TGS or SGS mode of operation is decided based upon dose rate and density measurements of a given waste drum. Canberra has

developed a special software package called the “Supervisor” for automatically deciding the assay mode (TGS or SGS) based on the density and dose rate measurements. Alternatively, the Supervisor software may also be used by the system’s operator to manually set an assay mode for drums in a given waste stream. A highly collimated ^{152}Eu source (15 mCi, 555 MBq) is used to perform the transmission scan.

A Canberra Model BEGE 5030 High Purity Germanium (HPGe) detector is used in the TGS system that has been built. The pulse processing electronics consists of an RC pre-amplifier, a Canberra Model 2060 Digital Signal Processor (DSP), and an Accuspec B Board to facilitate high speed data transfer, and a Canberra Model 1654 NIM Reference Pulser. Rate loss corrections were performed using the reference pulser counts. Data acquisition and analysis was performed using Canberra’s NDA2000 software platform.

Some of the salient features of the next generation TGS are discussed in the following sections.

Figure 1. The Next Generation TGS with fully automated features

The local PLC control pedestal is to the right and the transmission source assembly is to the left. A 300-liter drum is shown on the rotator/translator. The attenuator mechanism is visible in front of the HPGe detector. Between the attenuator(s) and the dewar is located the variable aperture collimator. The dosimeter is mounted on the front of the detector lift and is therefore, not visible in the picture.

When the system is installed at a waste assay facility, the in-feed and out-feed conveyor stubs shown in the picture can be readily integrated with an automated conveyor system that loads and unloads a stream of waste drums.

The sequence of events is as follows. A waste drum is loaded on to a conveyor using a crane or a fork lift. Upon receiving the signal from the PLC, the drum moves along the conveyor and eventually reaches the in-feed portion of the TGS conveyor. The drum is then transferred to a palette centrally located on a translator/rotator platform. The drum is weighed. The platform then moves towards the center-line of the detector-transmission source. The palette is raised up so that the bottom of the drum clears the side rails of the conveyor. The drum is then rotated. The dosimeter located at the front of the detector lift measures the average and maximum dose rates of the drum contents. Based on the bulk density of the drum matrix and the maximum dose rate, the Supervisor software selects either the TGS or the SGS assay mode. Once the assay mode has been selected, the NDA2000 software proceeds to set the geometry which is a combination

of the detector distance, collimator aperture, and the presence or absence of one or more lead attenuators. After the geometry is set, data acquisition is performed followed by data analysis and printing of the reports. In the case of TGS mode, the report includes images of the source distribution.

2.1 Detector Shield and Automated Collimator Assembly

The detector shield minimizes the impact of background gamma radiation and surrounds the sides and extends behind the detector. The shield provided is a minimum of 50 mm thick lead with a low-Z inner liner to minimize the effects of the lead X-rays on the measurement. The collimating aperture consists of lead and tungsten components. The aperture is formed by interleaved layers of tungsten that can be opened and closed to change either the spatial resolution of the TGS measurement or to switch over to an SGS measurement mode. Closing the tungsten leaves forms a diamond shaped aperture from 12.7 mm to approximately 60 mm in diameter. For SGS mode the aperture opens to a height of approximately 95 mm and width of 400 mm. The width of the opening is controlled via the system PLC and operating software.

2.2 Automatic Detector Slide Mechanism

The detector slide mechanism allows the detector to move towards or away from the sample rotator. The slide mechanism provide for a travel of +/- 355 mm from the normal detector position. The slide mechanism is automated to allow the detector to move further back from the high activity drums extending the dynamic range of the system.

For use with small samples such as cans, the can can be placed on sample pedestal to allow the detector to move closer to the container for improved spatial resolution in either the SGS or TGS modes.

2.3 Automated Attenuator Assembly

The automated attenuator assembly is intended to allow system operation for drums with high surface exposure rates. The assembly is mounted to the front face of the detector shield and collimating aperture. The assembly is controlled by the system PLC with the appropriate thickness of attenuator selected based on the measured dose rate for the sample drum.

The attenuator consists of 4 individual attenuators, 3 of lead and 1 of low-Z material. The inner most attenuator is the low-Z attenuator shutter. This shutter is included primarily to shield the detector from the lead X-rays generated in the three lead shutters. The lead attenuating shutters have thicknesses of 9, 8, and 7 mm respectively. Prior to the beginning of an assay, all of the lead attenuators are closed, thus protecting the detector from a potential high activity drum. Depending on the measured dose rate for a given sample drum, the lead attenuating shutters will be opened sequentially. The collimator aperture and the horizontal position of the detector are also automatically adjusted based on the measured dose rate.

These attenuator assembly can be used in either the SGS or TGS modes of operation.

3. Performance Test Results and Discussion

The TGS system was calibrated by placing a set of six mixed gamma rod source standards inside an empty 208-liter drum and performing a TGS assay. The rod sources were inserted at specific radial positions in the drum, with the radii ranging from 50mm to 280mm. The radial positions were arranged in a spiraling fashion. The rod sources consisted of the nuclides ^{133}Ba, ^{137}Cs, and

^{60}Co. The calibration results included in this paper are for the assay geometry where the TGS collimator aperture is set to a value of 50.3 mm, the detector horizontal position is set at the mid-position of the +/- 355.6 mm range, and all of the attenuators are open.

The output of a given TGS analysis is a quantity known as the "TGS Number" and the uncertainty associated with it. The TGS number and its uncertainty are determined at each emission energy, and represents values proportional to the activity or mass of an assayed radionuclide inside the drum. For a calibration count, the activity of the nuclides in the standard sources are known from the certificates. The calibration parameter is then determined as Activity per unit TGS number for a specific nuclide and a specific gamma ray of that nuclide. The raw TGS numbers for empty drum calibration measurement are given in Table 1.

Table 1. TGS Calibration Results (1 μCi = 3.7x10^4 Bq)

Gamma Energy (keV)	TGS Number	Uncertainty	μCi per TGS Number	Uncertainty
276	0.9722	0.0166	191.188	6.646
303	2.1725	0.0233	85.554	2.750
356	6.3186	0.0470	29.416	0.918
383	0.8815	0.0103	210.858	6.844
662	0.9407	0.0125	34.228	1.186
1173	0.7319	0.0143	42.226	1.619
1332	0.6571	0.0147	47.037	1.876

Several measurements were performed to verify the accuracy of the TGS calibration. Two such measurements are discussed below.

3.1 Uniform activity distribution inside a homogeneous matrix

In this assay, the six rod sources were inserted in radial holes drilled through a Homosote matrix (0.43 g.cm^{-3}) drum. The assay was performed for a total time of 1 hour. The assay results are given in Table 2. The transmission and emission images for ^{137}Cs and ^{60}Co gamma rays are given in Figure 2 and 3.

Table 2. TGS Accuracy for a Uniform Activity Distribution

Nuclide	Measured Activity		True Activity		Ratio (Meas/True)	
	Activity (μCi)	Uncert (μCi)	Activity (μCi)	Uncert (μCi)	Ratio	Uncert
Cs-137	33.18	1.23	32.04	1.03	1.036	0.051
Co-60	31.04	0.90	30.05	0.99	1.033	0.045

For all of the results presented in this paper, the uncertainty values are quoted at 1σ standard deviation. In each case, the measured uncertainty includes the statistical component evaluated using the Monte Carlo Randomization (MCR) method[4]. The measured activities are in excellent agreement with the true activities of the nuclides.

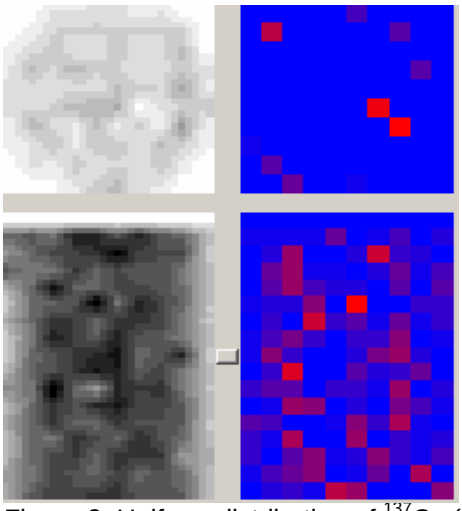

Figure 2. Uniform distribution of ^{137}Cs (662 keV)

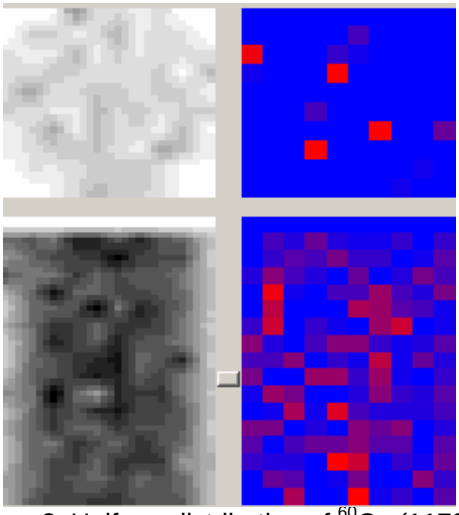

Figure 3. Uniform distribution of ^{60}Co (1173 keV)

The images on the left (in black and white) are the transmission images, and the images on the right (in colour) are the emission images. The transmission images shown are a 2-dimensional projection of the linear attenuation coefficient map (μ) at the given gamma ray energy. The emission images are a 2-dimensional projection of a voxel by voxel distribution of the nuclide activity. In figures 2 and 3, the transmission images show that the matrix is a homogeneous matrix. The emission images indicate the presence of nuclide activity throughout the drum volume, which is consistent with a uniform distribution of activity. In each figure, the top portion is the cross-sectional view of a given layer (layer 8 in Figures 2 and 3).

3.2 Three Point Sources of ^{137}Cs distributed in a homogeneous matrix

A TGS assay was performed by distributing 3 point sources of ^{137}Cs of approximately same activities inside a 208 liter Homosote matrix (0.43 g.cm^{-3}) drum. The results are given in Table 3. The transmission and emission images are shown in Figure 4.

Table 3. TGS Accuracy for 3 point sources inside a Homogeneous Matrix

Nuclide	Measured Activity		True Activity		Ratio (Meas/True)	
	Activity (μCi)	Uncert (μCi)	Activity (μCi)	Uncert (μCi)	Ratio	Uncert
Cs-137	187.5	6.56	178.40	5.35	1.051	0.048

The measured activities are in excellent agreement with the true activities at the 1σ level of uncertainties.

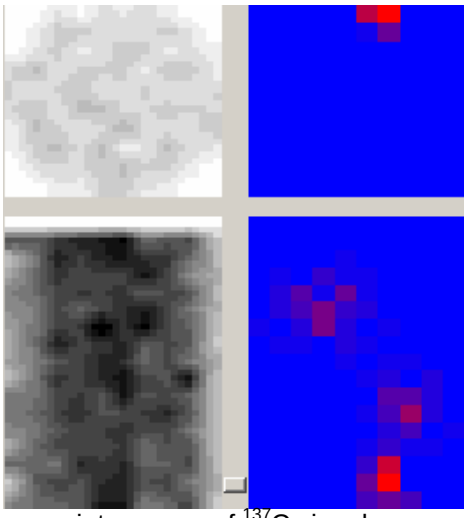

Figure 4. Three point sources of ^{137}Cs in a homogeneous matrix

The transmission image shows the uniform nature of the Homosote matrix. The emission image shows the presence of three regions with elevated levels of source activity. The source at the interface of layers 1 and 2 towards the edge of the drum is clearly seen in the cross-sectional view of the emission image.

3.3 Single Point Source of ^{137}Cs inside a Non-Homogeneous Matrix

A single point source is located inside a non-homogeneous matrix represents a difficult assay situation, with regards to the accuracy of the results. For such situations, the assay results from a non-imaging method might suffer from significant biases (for example a factor of 2). This is especially true if the matrix density is high. For an imaging method such as the TGS technique, the assay results might be more accurate. To explore this, TGS assays were performed by locating a single point source inside non-homogeneous drum matrices with moderate to high densities. The example discussed in this paper is one such typical assay of a ^{137}Cs point source inside a scrap steel drum with a bulk density of 1.0 g.cm^{-3}. The source was located at a radius of 220 mm and at a height of 390mm from the bottom of the drum.

Table 4. Single point source of ^{137}Cs inside a Non-Homogeneous Matrix (1.0 g.cm^{-3})

Nuclide	Measured Activity		True Activity		Ratio (Meas/True)	
	Activity (μCi)	Uncert (μCi)	Activity (μCi)	Uncert (μCi)	Ratio	Uncert
Cs-137	132.1	4.62	108.43	3.25	1.218	0.037

This is a difficult assay scenario from many respects. There is only a single point source, the matrix is non-homogeneous, and the density is fairly high. Considering these mitigating factors, the accuracy of TGS seems fairly reasonable. The images are shown in Figure 5.

In view of the encouraging results that we have obtained for a matrix density of 1.0 g.cm^{-3} we are exploring a variety of analysis methods to further extend the density range of TGS assays. These methods include the “Uniform Layer” and the “Bulk Density” approaches. In the Uniform Layer approach, the linear attenuation coefficient μ is calculated for a few voxels in a given drum layer for which transmission data is available and then averaged. All of the voxels of the given layer are then populated with this average value of μ. The Uniform Layer approach will be useful in assay

situations where the transmission beam is too weak to penetrate the drum matrix, while the drum contents are active enough to register non-negligible count rates in TGS views. In the "Bulk Density" approach, all the voxels in all the drum layers are populated with a value of μ derived based on the bulk density, assuming a matrix composition. This will be useful for drum matrices with very high densities (2 to 3 g. cm^{-3}). The Bulk Density approach will be useful in situations where the operator may only have TGS data available for a high density drum. Also, Bulk Density TGS assays generate confirmatory emission images that will be useful in assaying conditioned waste.

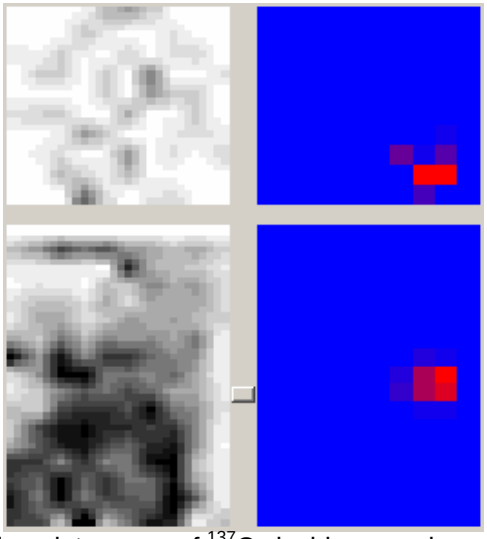

Figure 5. A single point source of ^{137}Cs inside a non-homogeneous matrix

The transmission image depicts the non-homogeneous nature of the scrap steel matrix. The emission image shows the presence of the single point source inside the drum. The image is remarkably constrained within four neighbouring voxels.

4. Conclusions

Canberra Industries has built and tested four fully automated enhanced TGS systems for waste assay. The salient features of this next generation of TGS systems were discussed. The performance of the system was evaluated using results from assays involving homogeneous and non-homogeneous matrices, and uniform and non-uniform source distributions. The design provides qualitative gamma assay capability over the surface dose rate range of up to 100 R/h (1 Sv/hr) and over the density range up to about 3 $g.cm^{-3}$ in a fully automated industrial environment. The TMU estimates for the TGS system are being evaluated and will be reported elsewhere in the near future.

5. References

1. S. Croft, TD Anderson, R.J. Estep, R.J. Huckins, D.L. Petroka, R. Venkataraman and M. Villani, *A new drum tomographic gamma scanning system*, Presented at the 25th ESARDA Symposium on Safeguards and Nuclear Material Management, Stockholm, Sweden, 13-15 May 2003.
2. S. Croft, R. Venkataraman, and M. Villani, *Characterizing a Tomographic Gamma Scanner*, Proceedings of 45th Annual INMM Meeting, Orlando, FL, July 18-22, 2004.
3. R. Venkataraman, S.Croft, M. Villani, and R.J. Estep, *Performance Study of the Tomographic Gamma Scanner for the Radioassay of drums,* Proceedings of 45th Annual INMM Meeting, Orlando, FL, July 18-22, 2004.

4. R.J. Estep, D.Miko, and S. Melton, Monte Carlo error estimation applied to Nondestructive Assay methods, Nondestructive Assay Waste Characterization Conference, Salt Lake City, Utah, May 22-26, 2000.

The Use of ^{252}Cf for Calibrating Safeguards Monitors

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Abstract:

Spontaneous fission ^{252}Cf neutron sources are commonly used to calibrate and test the operating performance of safeguards instruments and portal monitors. ^{252}Cf is preferred over (α,n) sources for many operational and safety reasons, but most importantly can be designed to emit an energy-dependent spectrum that closely matches that of ^{240}Pu. ^{240}Pu is the pre-dominant neutron emitter of plutonium metal and oxides. Because portal monitors and other border monitoring devices are designed to detect low neutron fluence rates at distances of 1-3 metres from the detector, several characteristics in the manufacture and decay of the source need to be understood and accounted for in order to make an accurate measurement of the performance: isotopic composition and ^{250}Cf to ^{252}Cf evolution, age, traceability, angular dependence of the neutron flux, and design and manufacture of the moderating cover are examples. This paper describes these issues and presents the range of possible error terms in the utilization of these physically small, relatively weak, (10^4 n.s^{-1}) neutron sources for calibration of portal monitors.

Keywords: ^{252}Cf; calibration; californium; plutonium

1. Introduction

For calibration of neutron coincidence counters and multiplicity counters, ^{252}Cf is used as a surrogate for ^{240}Pu, the principle isotope of interest in either low or high burnup plutonium fuels. In 1993 Croft [1] presented the results of Harwell Laboratory's work with ^{252}Cf as a calibration source for passive neutron counters. His work presents a comprehensive review of the issues and effects observed, in time, with the neutron multiplicity distribution, mean neutron energy, the Reals to Totals ratio, and the overall effect of source isotopic composition, particularly as it influences the neutron output over time. In passive neutron counters it is imperative to understand and account for these effects so that an accurate representation of the counter's performance relative to a mixture of spontaneously fissioning isotopes found in safeguarded and waste materials of plutonium (^{238}Pu/^{240}Pu/^{242}Pu).

In the last few years, a significant effort is underway for a large-scale development and deployment of homeland defense and border monitoring equipment. Two types of neutron detection systems have been deployed: large, fixed-geometry portal monitors and hand-held radioisotope identifiers (RIDs). Most of these systems use moderated ^{3}He proportional counters to detect fission neutrons. Various algorithms are used to analyze the signal neutron pulse train and alarm when the rate is determined to be greater than the background rate. ^{252}Cf sources are used to calibrate the systems for efficiency and to test overall system performance. Performance-based testing and calibration with ^{252}Cf provides assurances to the instrument users that IAEA Significant Quantities (SQs)[1] can be effectively detected and interdicted, as necessary.

[1] As defined under the terms of the NPT (Non-Proliferation Treaty, 1968), an IAEA significant quantity is "the approximate quantity of nuclear material in respect of which, taking into account any conversion process involved, the possibility of manufacturing a nuclear explosive device cannot be excluded."

This paper discusses the calibration issues of ^{252}Cf relative to the performance testing of these devices for international safeguards and interdiction, and presents an additional framework to the original efforts of McGarry and Boswell. [2]

2. ^{252}Cf Source Characteristics

In the testing of instruments, the nominal neutron count rate specified is between 10^4 and $2x10^4$ neutrons sec^{-1} $(4\pi\ Sr)^{-1}$ [see Section 3].

2.1. Isotope Atom Fractions

The isotopic composition of a ^{252}Cf source changes from target batch to target batch. While ^{252}Cf is produced in two facilities world-wide, we focus on the production facility at Oak Ridge National Laboratory, where over 95% of the targets are irradiated. [3], [4]. The isotopic composition of recent campaigns varies only slightly as shown in Table 1.

	Batch Code						
Cf Isotope	CXCF-598	CXCF-669	CXCF-579	CXCF-561	CXCF-537	CXCF-531	CAMP68
^{249}Cf	4.32	1.87	6.7	6.72	4.72	5.84	5.76
^{250}Cf	10.82	9.1	9.63	9.36	10.19	10.01	9.22
^{251}Cf	3.31	2.78	2.97	2.91	3.04	3.01	2.85
^{252}Cf	81.5	86.13	80.63	80.98	82.04	81.12	81.99
^{253}Cf	0.04	0.0005	0.03	0.02	0.01	0.01	0.165
^{254}Cf	0.01	0.008	0.04	0.01	0.01	0.01	0.018

Table 1. Atom Fractions (%) from ORNL., (1998-2002, Martin R.)

The atom fractions of the principle spontaneous fissioning isotopes of ^{252}Cf and ^{250}Cf are fairly constant between recent campaigns, as shown in Table 1. However, when targets of a different origin were used in the 1980s and early 1990s, the atom fractions were substantially different (particularly for the Russian-produced material). In fact, it was the case during testing of our portal monitors at an off-site facility, that a source from the 1980s was used and inaccurately 'calibrated' to the test date of interest. As the source ages it is very important to account for the neutron output of the ^{250}Cf isotope as well, as shown later in this paper.

2.2. Decay Properties of the Source

Literature values for isotopic properties are found in Table 2. The value for ^{251}Cf is estimated from systematics.

Cf Isotope	Half-Life (y)	Spontaneous Fission Branching Fraction
^{249}Cf	350.6	5.2×10^{-9}
^{250}Cf	13.08	7.7×10^{-4}
^{251}Cf	898	9×10^{-6}
^{252}Cf	2.645	3.1×10^{-2}
^{253}Cf	0.0515	-
^{254}Cf	0.165	9.97×10^{-1}

Table 2. Decay Properties of Constituent Isotopes.

3. Neutron Testing Requirements

A number of new test plans and requirements have been published in the last year, either by the IAEA, the IEC, or ANSI. We focus on ANSI here, American National Standards Institute, standards N42.34 and N42.35. [5], [6].

For RIDs (ANSI N42.34), the specification (§8.3.3) calls for the use of an un-moderated ^{252}Cf source of 0.01 µg. The source when placed 250 mm from the instrument must activate an alarm within 2 seconds (no repeatability tests required). At this distance from the instrument, the source produces a neutron dose rate of "approximately 0.3 mrem h^{-1}," the standard indicates (where 1 mrem is approximately 0.1µSv). No specific neutron output is provided nor is a requirement on the uncertainty of the source output. The assumption is that the source used is always "fresh" and free of other Cf isotopes and that it is "NIST traceable. This is often not the case and therein lies the problem.

For portals (ANSI N42.35), the specification calls for a ^{252}Cf source with an intensity of $2x10^4$ neutrons sec^{-1} $(4\pi\ Sr)^{-1}$ ± 20% (implied at 2 standard deviations from the mean). The source is used un-moderated, and is moved through the portal at a specified rate and distance. The alarm probability must be greater than or equal to 0.90 at a 95% confidence.

With a "fresh" source we see that each of the reference test cases specified uses a different source, given a nominal neutron yield 2.34×10^6 neutrons sec^{-1} $(4\pi\ Sr)^{-1}$ per µg ^{252}Cf. The RIDs use 2.34×10^4 n/s while the portals, 2×10^4 n/s, a 17% difference that creates an operational inconvenience to a test laboratory or during calibration – one would hope the same source would satisfy each of the requirements. In addition, the explicit details of source certification and expression of uncertainty should be given for these test sources, especially at such low yields, where the source cross-calibration techniques described by McGarry and Boswell [2] are pushed by today's practices. Simply, it is very difficult to produce certified reference standards at these rates with desired levels of precision. The ANSI test procedures do no allow for variations in excess of ± 20% (2σ), nor do they allow for decay of the source with time. The standards are essentially written such that a test, with Cf, can be conducted over a short period of time (e.g. months).

4. Source Certification

We present this information here not to question the quality of the manufacturer's work, but rather to give the end user of the source some thoughts to consider when actually using the source and applying it to "certification" test results. As we understand it, californium source suppliers normally produce a "source certificate" in one of two ways, though discussions and improvements are underway as we speak.

The default is to send a certificate showing that the source neutron yield is nominal. When a 20,000 $n.s^{-1}$ source is ordered, for example, the source certificate states that the nominal yield is 20,000 $n.s^{-1}$ at a 10% nominal uncertainty. The source is actually yielding between 18,000 $n.s^{-1}$ and 22,000 $n.s^{-1}$, with a Poisson variation of roughly 140 $n.s^{-1}$ (1σ). This is a very large uncertainty to deal with, given the conditions of the test. In portal monitors, for example, where the probability of an alarm must satisfy the 0.90 probability at a 95% confidence, a source yielding 18,000 $n.s^{-1}$ will either fail, or the system is over-designed (and therefore possibly too costly).
The second way is for the manufacturer to certify the source against a secondary reference material. The customer, for example, receives the same source as in the "default" case, but the certificate reports the actual neutron yield measured. That is to say, instead of 20,000 $n.s^{-1}$, the certificate may read 19,137 $n.s^{-1}$. The real capability to claim better than 10% uncertainty on the (inter-calibration) measurement is probably pushing the limit of the measurement device used by typical source vendors---again this is being evaluated.

In either case, we have had the traditional luxury of performing very accurate self-calibration measurements in our neutron multiplicity counters (using for example the Large Efficient Multiplicity Counter or LEMC) and we will present those results within the next year. As a result, we've been able to predict quite accurately what the actual neutron yield is. Others have not had this luxury.

5. Decay Corrections

The first and most significant judgement error that has been made is using sources that have been "laying around" in the back vault for years. Normally, the end user will decay correct the source activity assuming only ^{252}Cf is present. For sources older than 7-8 years, this calculation will underestimate the neutron yield because it inadequately accounts for the ^{250}Cf contribution, which decays much more slowly (13.08 y half-life).

The proper formalism for treating the neutron yield, with time is to account for the ^{250}Cf, in conjunction with the more rapidly decaying isotope, ^{252}Cf:

$$S(t_2) = S(t_1)e^{-\lambda_{252}(\delta t)}\left[\frac{1 + R_1 e^{\lambda_{eff}(\delta t)}}{1 + R_1}\right]$$

$S(t_n) = \textit{neutron rate at time, } t_n$

$\delta t = t_2 - t_1$

$\lambda_{252} = \textit{decay const } {}^{252}Cf$

$\lambda_{250} = \textit{decay const } {}^{250}Cf$

$\lambda_{eff} = \lambda_{252} - \lambda_{250}$

$R_1 = S_{250}(t_1) / S_{252}(t_1) \qquad R_1 = 0.004742$

6. Summary

In the science of calibrating passive neutron counters, we have been able to make very accurate measurements and predictions of the source yield as a function of decay time because we measure directly the single, doubles, and triples rate using multiplicity counting methods, and, have had access to well characterised reference materials. As a result, we've not been that insistent on the manufacturer's to tighten up their uncertainty estimates of neutron yield; furthermore, the neutron rates have been high enough to exceed statistical issues that are encountered with sources of 10^4 n.s^{-1}. However, in the case of calibrating and testing the newly developed homeland defense devices, the low-level count rate and the fact that we tend to use old sources for testing, creates new problems when certifying the instruments. Greater attention is needed on the issue of source certification, expressions of uncertainty, and instrument performance. In the case where old sources are used, source decay should be performed correctly, as presented here.

7. Acknowledgements

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8. References n.s^{-1}

[1] Croft S; Wakefield G; Chard P; Lambert K; Lloyd D; *Topics in Neutron Multiplicity Counting at Harwell*; proceedings of the 1993 Workshop on Passive Neutron Coincidence Counting; 1993.

[2] McGarry E; Boswell E; *Neutron Source Strength Calibrations*; NBS Special Publication SP 250-18; US Department of Commerce; National Bureau of Standards; March 1988.

[3] Osborne-Lee I; Alexander C; *CALIFORNIUM-252: A Remarkable Versatile Radioisotope*; ORNL/TM-12706; 1995.

[4] Martin R.; Knauer J; Balo P; *Production, Distribution, and Applications of Californium-252 Neutron Sources;* Oak Ridge National Laboratory presentation to IRRMA 1999, 4th topical Meeting on Industrial Radiation and Radioisotope Measurement Applications; 1999.
[5] ANSI N42.34-2003; *American National Standard Criteria for Hand-held Instruments for Detection and Identification of Radionuclides*; 2003
[6] ANSI N42.35-2003; *American National Standard for Evaluation and Performance of Radiation Detection Portal Monitors for Use in Homeland Security*; 2003.
[7] Reilly D; Ensslin N; Smith H; *Passive Nondestructive Assay of Nuclear Materials*; NUREG/CR-5550; 1991.

Characterisation of PuBe sources by Non Destructive Assay

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Abstract:

In this paper we describe the characterisation of PuBe, AmBe and AmLi neutron sources based on a combination of Non Destructive Assay (NDA). The sources were selected to cover the Pu mass range and the typical isotopic composition of the total inventory of sources in Hungary. The nuclide composition, thermal power, and neutron emission rate are deduced from a combination of gamma, calorimetric and neutron measurements. All measurements are performed with instruments previously calibrated using certified reference material which are traceable to primary standards. The radionuclide distribution is obtained from the ratio of full-energy net peak areas observed in the gamma spectra recorded by a high resolution Ge detector. The performance of a calorimeter based on thermopile technology is verified with experimental data obtained in in-field measurement conditions. From the radionuclide distribution together with the specific thermal power, the total mass of Pu and Am is deduced. We also report the measurements of the neutron emission rate obtained with a neutron counter calibrated with neutron sources for which the neutron emission is traceable to the primary standards of the National Physics Laboratory (NPL) of the United Kingdom. Therefore, the neutron sources characterised in this campaign can be used as secondary standards with a neutron emission rate traceable to primary standards at NPL. The values of the neutron emission rate, obtained with the neutron counter, are compared to the values deduced from the thermal power and the specific neutron emission rate deduced from the results of gamma measurements and nuclear data. This comparison is used to deduce a specific neutron emission rate value, which is representative for the neutron sources which have to be characterised.

Keywords: NDA; PuBe, AmBe, AmLi neutron sources; calorimetric measurements; neutron counting; illicit trafficking; safeguards; nuclear forensic analysis

1. Introduction

A large number of sealed (α,n) sources existing in the European Union represent a potential risk of illicit use. Hungary has about 250 AmBe and 200 PuBe sources which are mostly out of use. These sources are stored temporarily in the Institute of Isotopes, Chemical Research Centre (IICRC) of the Hungarian Academy of Sciences. The sources contain nuclear material (fissile and/or fertile) and are therefore under nuclear safety and safeguards control. Their content of nuclear material is to be accounted for and reported to the IAEA, and from 2004 onwards also to the

EURATOM safeguards authorities. Most of the sources were supplied by the former Soviet Union. In the source records only the source activity, the neutron output, and the specific neutron yield is declared. Therefore, the quantity of nuclear material is not directly documented, and the quantity of Pu and Am can only be determined indirectly from a combination of the neutron output and the specific neutron yield. In order to verify the declarations of the supplier, the IICRC initiated a series of independent neutron and gamma measurements [1]. This measurement campaign was strongly supported by the Hungarian Atomic Energy Authority and the IAEA. A number of PuBe sources, with declared Pu mass ranging from 0.18 to 177 g, were measured. In some cases the determined Pu mass deviated by an order of magnitude from the declared value, which is still the basis of the bookkeeping and reporting to the IAEA and EURATOM safeguards authorities.

To support these activities and in order to clarify the discrepancies, the Neutron Physics Unit of the IRMM and the Nuclear Safeguards Unit of the IPSC initiated in 2004 a collaboration with the IICRC. As part of this collaboration a neutron detection system and a calorimeter were transported to the IICRC. The first neutron and calorimetric measurements started in December 2004. Both instruments had previously been calibrated in the PERLA laboratory of the IPSC using certified reference materials. The calibration procedures and the traceability of the results for the neutron counter and the calorimeter are discussed in Ref. [2] and [3], respectively.

2. Characterisation of neutron sources by NDA

To characterise the neutron sources, we followed the measurement procedures described by Ravazzani et al. [4]. In Ref. [4] a set of AmBe, AmB, AmF and AmLi were characterised based on a combination of gamma spectroscopy, calorimetric and neutron measurements. The experimental data result in a direct determination of the nuclide composition, the thermal power and neutron emission rate. When combining gamma and calorimetric measurement results, the total quantity of Pu and Am contained in (α,n) sources can be obtained. Unfortunately the calorimetric measurement of the thermal power requires a long measurement time. One of the objectives of the measurement campaign was to verify if reliable values for the amount of nuclear material can be deduced from a combination of the total neutron emission rate and the specific neutron yield.

2.1. Gamma-spectroscopy

Ref. 4 shows that the results of gamma spectroscopic measurements can be used both to identify the type of neutron source, and to determine the nuclide composition. From the ratio of the full-energy net peak areas, resulting from photons emitted by different radionuclides, the relative quantities (or activities) of these nuclides can be determined [5]. A large area planar detector was used for this purpose. The Pu isotopic composition and the ratio of the amount of ^{241}Am to the total amount of Pu was determined by analysis of the gamma spectrum using the Multi-Group Analysis code MGA++ [6]. The isotopic composition together with the nuclear data of Table 1 is used to calculate for each source: the specific activity, the specific thermal power and the specific (α,n) neutron yield.

Nuclide	Specific activity [7] (GBq/g)	Specific thermal power [8] (mW/g)	Specific neutron yield [9] (s^{-1} Bq^{-1})
^{238}Pu	633.40	567.57	8×10^{-5}
^{239}Pu	2.296	1.9288	6×10^{-5}
^{240}Pu	8.398	7.0824	6×10^{-5}
^{241}Pu	3811.0	3.4120	
^{242}Pu	0.145	0.1159	
^{241}Am	126.84	114.20	8×10^{-5}

Table 1: The specific activity, specific thermal power and specific (α,n) neutron yield for ^{241}Am and Pu-isotopes. The (α,n) neutron yield of ^{241}Pu and ^{242}Pu can be neglected in the calculation of the specific neutron yield.

Also the total quantity (or activity) of radionuclides can be determined from the gamma spectroscopic measurements. This requires a correction for the attenuation of gamma rays in the source and the encapsulation. In Ref. 4 the infinite energy method, originally proposed by Morel [10], is used to correct for the attenuation of the gamma rays. At the IICRC another method is proposed which uses the ratio of gamma lines to determine the energy differential attenuation correction factors. The details of these measurements and the data analysis procedures are discussed in another contribution to this symposium.

2.2. Calorimetric measurements

The Small Sample Calorimeter (SSCAL), developed by ANTECH of Wallingford, Oxfordshire, UK, is one of a new generation of calorimeters that employ thermopile technology in conjunction with traditional nickel resistance thermometry. Thermopiles consist of a serially interconnected array of thermocouples, which produces an offset free Seebeck voltage, proportional to the temperature difference between two sites. The SSCAL is a heat flow calorimeter which measure the voltage generated across a thermal gap (consisting of a thermopile cup) when it contains a heat-emitting sample [3]. Different control regimes are possible in the SSCAL due to the implementation of both thermopiles between the block and the outer cylinder, and nickel sense winding on the block and the outer cylinder. In the present measurement campaign we used the following two control mechanisms: nickel sensor to control the temperature on the outer cylinder, and thermopile annulus to control the heat flow between the block and the outer cylinder.

Thermal equilibrium in a measurement is achieved when the outer cylinder is at constant temperature (a few degrees below the temperature of the block) and the heat flow rate between the block and the outer cylinder is constant. At equilibrium, the heat flow rate measured in the thermopiles of the sample cup is proportional to the sample power. Figure 1 shows the time required to reach equilibrium. A small sample 48 mW need about 2 hours while a sample of 240 μW needs about 8 hours.

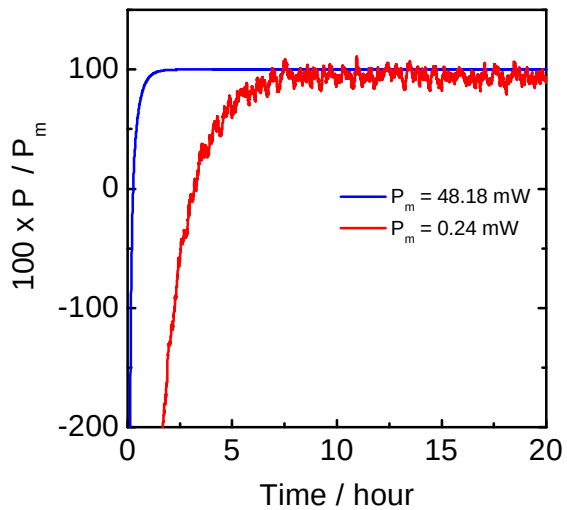

Figure 1: SSCAL measurement of sample power for two PuBe samples

The thermal power of each neutron source was measured with the SSCAL. The total thermal power of a source P_m, is a function of the total mass, the weight fraction w_j and the specific thermal power, p_j, of the heat producing radionuclides:

$$P_m = m\sum_j w_j p_j \qquad (1)$$

Therefore, the total mass can be determined by combining a thermal power measurement with a gamma spectroscopy measurement of the radionuclide distribution.

We verified the linearity of the calorimeter response using an electrical heat standard. Figure 2 shows the ratio of measured power to inserted electrical power at various selected values. These results indicate that the non-linearity in the range of 20 mW to 200 mW is less than 0.1%.

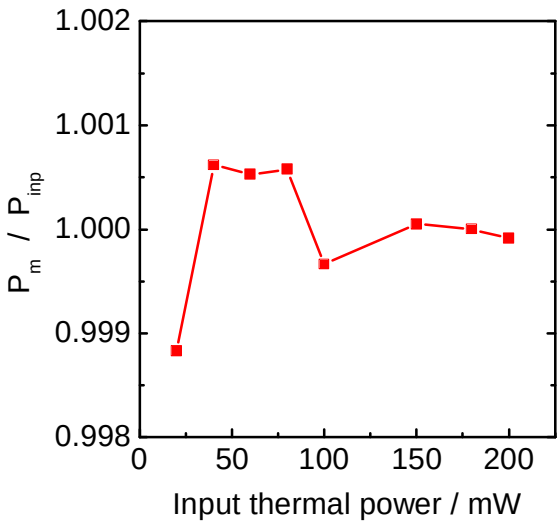

Figure 2: Measured to inserted power of electrical heat standard

Repetitive measurements of neutron sources with a thermal heat output ranging from 0.2 mW to 50 mW were carried out. Results are given in Figure 3. For each source the standard deviation of the measured power was determined from the variation in repeated measurement. The straight line in Figure 3, representing a constant standard deviation, reveals that the minimum uncertainty of the instrument is 0.05 mW independent of the thermal power of the source. This value results in a precision of 17% and 0.03% at 0.3 mW and 150 mW, respectively. According to the instrument manual the optimal operation range is between 0.2 and 50 mW with a precision better than 3.0% and 0.2% at 0.3mW and 150mW, respectively.

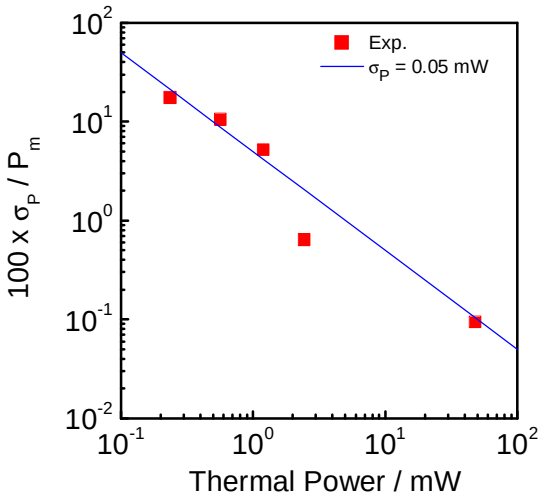

Figure 3: Relative standard deviation as function of the thermal power of the neutron source

2.3. Neutron measurements

The neutron source strength is obtained from measurements with a JCC–13 neutron counter developed by Canberra [11]. The detector is a passive well-counter incorporating 18 ^{3}He proportional counters in a polyethylene moderator. The measurement cavity is lined with a 1.2 mm thick cadmium sleeve. The detector was connected to a JRS-12 Shift Register. In this work we only report the total neutron count rate. Ref. [1] shows that also the small contribution of correlated neutrons can be used as a signature of the nuclear material in PuBe sources. The correlated events in PuBe sources result mainly from neutron induced fission reactions and the ^{9}Be(n,2n)^{8}Be reaction. The JCC-13 counter was calibrated with reference neutron sources, which have a declared total neutron emission rate that is traceable to the primary standard of the National Physics Laboratory (NPL) of the United Kingdom. The detection efficiency for AmLi and AmBe sources is 0.513 and 0.321, respectively. In Ref. 4 the detection efficiency for PuBe sources is not reported. Since the energy spectrum for AmBe and PuBe are similar, we use a detection efficiency for thr PuBe source neutrons equal to the value obtained for AmBe source neutrons.

When neglecting the self-multiplication, the total neutron emission rate $S_{\alpha n}$, is a function of the total mass m, the weight fraction w_j and the specific neutron emission rate $s_{\alpha n,j}$:

$$S_{\alpha n} = m \sum_j w_j s_{\alpha n,j} \qquad (2)$$

In the case the specific neutron emission rates can be considered nuclear constants (i.e. independent of the type of neutron source, the size of the neutron source and fabrication procedures), the total mass can be determined from a combination of the neutron emission rate and the radionuclide distribution obtained from gamma measurements. One of the objectives of this work is to verify if this is valid in practice.

3. Results and discussion

During the measurement campaign we characterised 13 PuBe, 4 AmBe and 1 AmLi neutron sources. The specific neutron emission rate and the thermal power are deduced from the nuclide distribution w_j obtained from the gamma measurements together with the nuclear data of Table 1. The Pu mass is derived from knowing the nuclide distribution and the total thermal power of the source.

The results for the AmBe and AmLi sources are given in Table 2. The specific neutron emission rates are compared with the data of Ref. 4 and the values recommend in the ISO-8525 standard [12]. Although the specific emission rates are in agreement with the values quoted in Ref. 4 and 12, the data in Table 2 indicate a significantly lower value for the sources measured at IICRC. This systematic difference could result from a difference in the production process.

Table 3 summarises the results from the gamma, calorimetric and neutron measurements for the PuBe sources. The total amount of Pu ranges from 23 mg to 45 g. The relative amount of ^{239}Pu is between 76 % and 96 %. The last column of Table 3 shows the ratio of the neutron emission rate obtained by neutron counting to the emission rate obtained from a combination of gamma spectroscopy and calorimetry using the nuclear data of Ref. 9. For sources with ^{239}Pu around 95 wt%, a relatively constant ratio of about 0.49 with a standard deviation of 2% is observed. For sources with ^{239}Pu < 90 wt% the ratio increases systematically with increasing relative ^{239}Pu content. This indicates that for these sources a representative specific neutron emission rate can be defined. Such a representative specific emission rate provides a method for determining the Pu mass of the

unknown PuBe sources with relatively good accuracy by a gamma measurement and a neutron counter measurement alone, thus avoiding the time consuming calorimetric measurement. Before defining the specific emission rate, the systematic behaviour in Table 3 as function of relative ^{239}Pu has to be investigated. This systematic behaviour could be influenced by both the detection efficiency and the nuclear data.

	Declared	JCC-13/14		SSCAL		IICRC	Ref. 7	ISO-8525 [12]
	$S_{n,d}$ (s^{-1})	$S_{n,n}$ (s^{-1})	$S_{n,n}$ / $S_{n,d}$	P_m (mW)	^{241}Am (mg)	$s_{n,Am}$ (s^{-1} g^{-1})	$s_{n,Am}$ (s^{-1} g^{-1})	$s_{n,Am}$ (s^{-1} g^{-1})
AmLi	54780	37820	0.69	37.93	332.1 ± 0.5	1.1×10^5	$1.5\ (0.3) \times 10^5$	1.4×10^5
AmBe (1)	84350	70250	0.83	1.20	10.5 ± 0.5	6.7×10^6		
AmBe (2)	233500	187500	0.80	2.74	24.0 ± 0.5	7.8×10^6	$8.8\ (0.8) \times 10^6$	8.4×10^6
AmBe (3)	681000	457100	0.67	7.89	69.1 ± 0.5	8.3×10^6		
						Av. = 7.6×10^6		
						St.dev = 0.8×10^6		

Table 2: The neutron emission rate $S_{n,n}$ deduced from measurements with the JCC-13 is compared to the declared neutron emission rate $S_{n,d}$. We also report the observed thermal power P_m and the resulting amount of ^{241}Am. The resulting specific neutron emission rate is compared to data in literature.

PuBe	Declared	Gamma			SSCAL	JCC-13					
	$S_{n,d}$ (s^{-1})	^{239}Pu (wt %)	$\Sigma w_j p_j$ (mW g^{-1})	$\Sigma w_j s_{nj}$ ($s^{-1} g^{-1}$)	P_m (mW)	$Pu_{p\gamma}$ (g)		$S_{n,p\gamma}$ (s^{-1})	$S_{n,n}$ (s^{-1})	$S_{n,n}$ / $S_{n,d}$	$S_{n,n}$ / $S_{n,p\gamma}$
497	5.56×10^6	76.2	11.40	9.92×10^5	122.04	10.39	± 0.32	1.06×10^7	4.61×10^6	0.83	0.43
458	5.50×10^5	78.4	10.60	9.18×10^5	13.39	1.23	± 0.04	1.16×10^6	5.18×10^5	0.94	0.45
428	5.00×10^5	79.9	10.92	9.48×10^5	12.39	1.11	± 0.03	1.08×10^6	5.47×10^5	1.09	0.51
603	3.14×10^5	88.2	6.24	5.19×10^5	6.37	1.01	± 0.03	5.30×10^5	2.88×10^5	0.92	0.54
479	1.16×10^5	79.8	10.76	9.30×10^5	2.46	0.223	± 0.008	2.12×10^5	1.01×10^5	0.87	0.48
480	1.10×10^4	83.2	10.41	9.00×10^5	0.24	0.022	± 0.005	2.05×10^4	1.06×10^4	0.96	0.52
701	5.27×10^6	94.9	2.48	1.82×10^5	109.96	44.29	± 1.33	8.09×10^6	4.51×10^6	0.86	0.56
407	2.68×10^5	95.2	2.44	1.79×10^5	6.41	2.62	± 0.08	4.70×10^5	2.71×10^5	1.01	0.58
625_2	2.66×10^5	95.3	2.43	1.79×10^5	6.34	2.60	± 0.08	4.66×10^5	2.78×10^5	1.04	0.60
625_3	2.56×10^5	95.5	2.44	1.79×10^5	6.10	2.50	± 0.08	4.49×10^5	2.60×10^5	1.02	0.58
611	1.16×10^5	95.7	2.46	1.81×10^5	2.68	1.09	± 0.04	1.98×10^5	1.13×10^5	0.97	0.57
601_1	1.00×10^5	95.8	2.38	1.75×10^5	2.46	1.03	± 0.04	1.81×10^5	1.07×10^5	1.07	0.59
601_2	1.00×10^5	95.6	2.45	1.80×10^5	2.39	0.97	± 0.04	1.76×10^5	1.01×10^5	1.01	0.58

Table 3: Assayed Pu mass ($Pu_{p\gamma}$) and comparison of declared ($S_{n,d}$) and measured neutron emission rates ($S_{n,n}$ and $S_{n,p\gamma}$). The measured rates are obtained from direct neutron measurements ($S_{n,n}$) or by combining gamma spectroscopy with calorimetric measurements ($S_{n,p\gamma}$) using the nuclear data of Ref. 9.

4. Conclusions

A series of PuBe neutron sources were assayed by three NDA methods: gamma spectroscopy, calorimetry and neutron counting. The sources were chosen to be representative, in radionuclide distribution and Pu mass, of the entire PuBe source inventory in Hungary. The Pu mass was determined and the original declaration of the neutron emission rate of the sources was confirmed by the measurements. The PuBe sources of 95% ^{239}Pu showed a consistent value of the specific neutron emission rate (per gram of sample). The PuBe sources of lower ^{239}Pu content (76 to 88 %) shows a relatively modest spread in the specific neutron emission rate. This spread however is subject for further investigation.

The result of this work indicates a way to characterise the remaining PuBe source inventory by gamma spectroscopy and neutron measurements alone. The gamma measurement characterises the radionuclide distribution, and thereby the conversion factor for the neutron measurement (table 3). The neutron measurement gives the neutron emission rate which determines the Pu mass when using the relation demonstrated in table 3.

5. References

[1] Bagi J., Tam Nguyen C., Lakosi L., "Assesment of the Pu content of Pu-Be neutron sources", Nucl. Instr. Meth., B222, 2004, p 242

[2] Schillebeeckx P., Farese N., Jaime R., Looman M., Pedersen B., Peerani P., Jackson K., Swinhoe M., Schwalbach P., Maiorov V., Thaurel B. and Weber A., "The PERLA Calibration Procedures for Passive and Active Neutron NDA Instruments", Proceedings of the 23rd Annual Symposium on Safeguards and Nuclear Material Management, Bruges, Belgium, 8 – 10 May 2001, p. 162

[3] Thornton M. and Schillebeeckx P., "Performance Evaluation of a Thermopile Based Small Sample Calorimeter in PERLA", Proceedings of the 23rd Annual Symposium on Safeguards and Nuclear Material Management, Bruges, Belgium, 8 – 10 May 2001, p. 174

[4] Ravazzani A., Jaime R., Looman M.R., Pedersen B., Peerani P., Schillebeeckx P., Thornton M.I, Foglio-Para A., Maiorov V.P., "Characterisation of Neutron Sources by NDA", Proceedings of the 23rd Annual Symposium on Safeguards and Nuclear Material Management, Bruges, Belgium, 8 – 10 May 2001, p. 181

[5] Gunnink R., "MGA: A Gamma-Ray Spectrum Analysis Code for Determining Plutonium Isotopic Abundances" , Lawrence Livermore National Laboratory report UCLR-LR-103220, Vol. 1, ISPO No. 317, Task No. A.161, April 1990.

[6] Multi-Group Analysis MGA++, Software User's Mnual, EG&G ORTEC.

[7] Lammer M. and Schwerer O., Handbook of Nuclear Data for Safeguards, IAEA INDC (NDS) – 248, June 1991.

[8] American National Standard for Nuclear Materials, Plutonium-Bearing Solids Calibration Techniques for Calorimeric Assay, ANSI 15.22 – 1987

[9] West D. and Sherwood A.C., Ann. Nucl. Energy, 9 (1982) p. 551

[10] J. Morel, M. Vallee, and M. Etcheverry, "Programme de mesure des quantités de plutonium par spectrométrie gamma Logiciel PuMa", Rapport CEA R-5482, Avril 1989

[11] Canberra product catalog, edition twelve

[12] "Neutron reference radiations for calibrating neutron-measuring devices used for radiation protection purposes and for determining their response as a function of neutron energy", ISO – 8529 (1989)

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ATOM (Accountancy and Tracking Of Material)

Today's nuclear industry is faced with wide-ranging regulatory requirements. It must not only comply with these requirements but must clearly demonstrate compliance. Foremost amongst these are those governing the movement, storage, and operations involving radioactive (RAM), and more specifically, nuclear material (NM). For example, of particular interest within the UK are the Environment Agency (or Scottish Environment Protection Agency), the Nuclear Installations Inspectorate, and the European Commission Directorate of Transport and Energy (formerly Euratom).

To meet the various regulatory requirements, and provide robust management controls on the quantities and whereabouts of materials held, a sophisticated tracking and reporting database is essential. UKAEA, in partnership with Fujitsu Services, have developed such a system, ATOM (**A**ccountancy and **T**racking **O**f **M**aterial). ATOM tracks all RAM and NM, through the various operations and locations, and produces the necessary regulatory reports, while maintaining a full audit trail of the changes. Its unique design allows for the reconstruction of the inventory at any given time, and can offer assistance with criticality safety arrangements and liabilities management for the ultimate return or disposal of the materials. This coupled with UKAEA expertise in meeting UK and International nuclear materials control requirements is a powerful combination in the field of RAM and NM management.

UKAEA are now offering a total solution to the challenges of materials management.

We will provide your company with an ATOM database system, built on ORACLE ™, delivered direct to your company desktop computers. The system will be tailored to your organisation's requirements, based on the Material Balance Areas defined by the Safeguards authorities, reflecting the physical location hierarchy of your facilities. We will map your buildings, and the areas within them, in which RAM and NM are stored and processed. This mapping can cover every conceivable operational area, from reactor cores to fuel storage ponds and racks. The result is a single, company-wide, solution offering uniformity and significant savings over supporting disparate systems with differing demands.

In addition to meeting the international safeguards requirements our service is of great benefit in meeting business needs such as planning and monitoring plant throughput, tracking materials against contracts, assisting with criticality and other safety arrangements, and managing the liability for treatment and disposal of materials.

Our solution also offers the advantages of no up-front capital costs, no requirement for your own full-time trained staff to run computer servers and manage the system, and no need to provide a helpdesk service. There is also no need to develop your own costly support arrangements to meet future business and regulatory needs. All this is provided as part of the UKAEA solution.

Exhibitor's session

Illicit Trafficking, Sub/cross-national threats, Poster presentation

A Comparison of an HPGe-based and NaI-based Radionuclide Identifier (RID) for Radioactive Materials

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Abstract:

The three steps in the interdiction process for illicit trafficking of radioactive and nuclear materials are detection, location and identification. Many handheld radioisotope identifiers (RIDs) have been introduced with the claim to the latter two tasks for gamma-emitting sources. Recent emphasis has been on identification in order to reduce the false positive rate. Previous work compared the search mode of an HPGe-based and a NaI-based RID for neutron and gamma-ray sources. In the present work, the performance of the same two instruments in the identification of radionuclides is compared. The requirements for the identification are given in ANSI N42.34 and related standards. The tests in the standard are for single nuclides and certain mixtures of nuclides for both shielded and unshielded sources. The two systems have been compared under the standard test conditions for a variety of the sources and configurations specified.

Results comparing the performance of both systems, with the standard and with each other will be given. The results show the HPGe system provides the correct identification more rapidly and on less spectral data than the NaI system and that it easily exceeds the ANSI N42.34 requirement.

Keywords: radioisotope; identifiers; germanium detectors; HPGe; illicit trafficking

1. Introduction

The Detective EX is a handheld radioisotope identifier (RID) based on a high purity germanium detector (HPGe) and moderated ^{3}He tubes for the gamma ray and neutron detection respectively. The size of the HPGe and neutron detectors was based on the efficiency requirements of ANSI N42.34 for the detection of differing amounts of material and on the ability to correctly identify the various nuclides in mixtures. The mixtures specified in the standard are those which could be used to hide prohibited material by masking it with other, innocent, radioactive materials. HPGe is arguably the only material available today which has both the necessary high resolution and adequate detection efficiency in a single detector. The neutron detectors were not used for these identifications.

Because of the relatively small size of the HPGe detector (about 12% relative efficiency), there is a widely held misconception that accurate nuclide identification will require a long counting time in order to obtain the necessary spectrum counts. However, these results show that the improved resolution of the HPGe more than compensates for the fewer overall counts than in the scintillator spectrum.

In the expected mode of operation of the RIDs, the location of the source is determined in the “search mode” and then the identification of the nuclides is determined in the identify mode. Prior work [1,2,3,4]

showed that high resolution gamma-ray detectors have the same or greater sensitivity than the most common size of low resolution detectors and that the HPGe detector systems can correctly perform the identification, even for mixtures, in less time than the requirements in the standard.

This work compares the time for correct analysis for the HPGe, high resolution RID with a scintillator based RID.

2. Experimental Configuration

The two RIDs were described earlier [1] and are shown in Fig. 1. The tests were conducted using the two instruments sequentially. The sources were placed in front of the instrument being tested at 10 cm from the front face of the housing (not necessarily the detector) and aligned with the active center of the detector as defined by the manufacturer. The sources were positioned with low mass and low atomic number materials to reduce scatter. The DU samples were positioned so that the maximum surface area was facing the detector. When the point sources were used with the DU, the point sources were positioned beside the DU to eliminate any shielding by the DU. The average dose, as recorded by the instruments, was about 50 ：Sv/h above background, consistent with the ANSI N42.34 [5] requirements.

Figure 1. The Detective and scintillator-based RIDs.

The Detective EX continuously collects and analyzes the spectrum. Once a second, the current analysis result is updated on the display. The running live and real time are also displayed. For each source, the identification was started with the source in place and the screen was continuously observed so that when the correct result was displayed, the real time (representing the data collection time) could be recorded. The identification was repeated and the results shown are the average of the multiple analyses.

The scintillator-based RID operates by collecting a spectrum for a fixed time, storing the spectrum, analyzing the spectrum and displaying the result. For this test, the collection time was set to the minimum time and the manual time extension mode was used to extend the time of the spectrum collection until a result was observed. The recorded times are only for the spectrum collection and do not include the extra time for processing, typically about 25 seconds

Both instruments were operated at the same confidence level on the peak areas for the peak to be recognized as present in the spectrum. This level was set such that the uncertainty of the net peak area was less than 5 sigma for most peaks. This level ensures a low level of false positives.

For either instrument, if the correct analysis was not observed in 180 seconds, the process was stopped and the result recorded as a failure. The maximum counting time is not given in ANSI N42.34, but is listed in the IAEA draft TECDOC [6] as 60 seconds for unshielded sources.

Some of the sources were point sources and some were extended sources. The list of sources is shown in Table 1.

Source	Nominal activity	Form	Correct report
Cs 137	1 uCi	Point source	Cs 137
Co 60	1 uCi	Point source	Co 60
Ba 133	1 uCi	Point source	Ba 133
Depleted uranium	98 g	Bar (5 x 5 x 30)	Depleted uranium
Depleted uranium	18 g	Thin plate	Depleted uranium

Mixed 1888	0.5 uCi each	Point source	Am241, Cs137, Co60, Cd109
Thorium	200 g natural thorium	Bulk powder, about 20 mm x 15 mm diameter	Th 232, Th 228, or Ra 226
Natural Uranium	Unknown	Uranium in glass	Uranium
Natural Uranium	5000 bq	Compressed powder, 5 mm x 40 mm diameter	Uranium

2.1. Example spectra

As soon as the correct analysis was observed in the Detective, the spectrum was manually saved. An example spectrum for identifying ^{60}Co is shown in Fig. 2. In this spectrum, note the relatively low number of counts needed to make the identification.

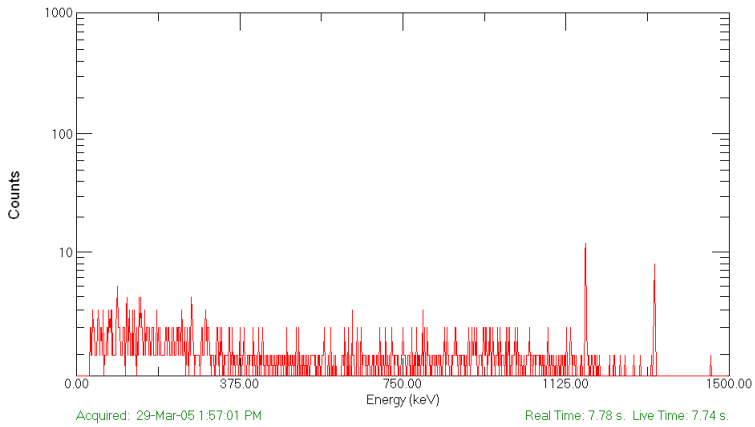

Figure 2. Short time spectrum of ^{60}Co from HPGe detector.

The DU and ^{133}Ba combination is shown in Fig. 3. In this spectrum, note the closely spaced gamma-ray peaks in the spectrum, which are easily separated by the HPGe detector.

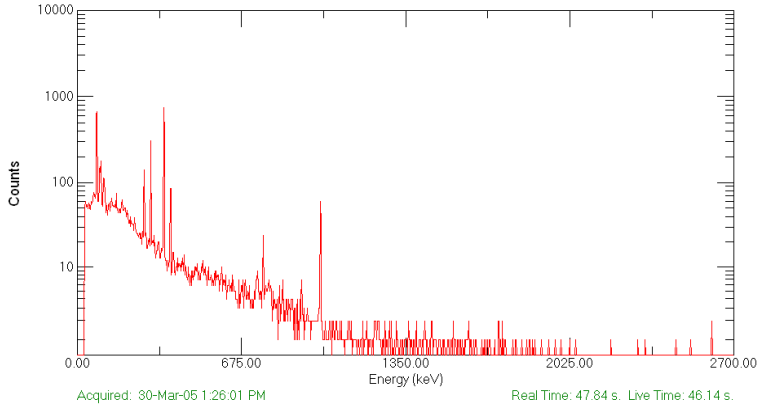

Figure 3. HPGe Spectrum of depleted uranium and ^{133}Ba.

The corresponding spectra for the scintillator system are shown in Figs. 4 and 5.

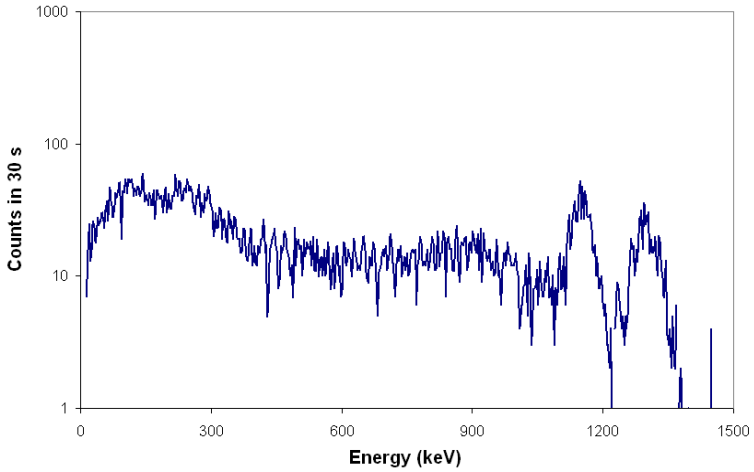

Figure 4. Scintillator spectrum of ^{60}Co.

The scintillator spectrum does not show the separation of the gamma-ray lines, making the identification difficult.

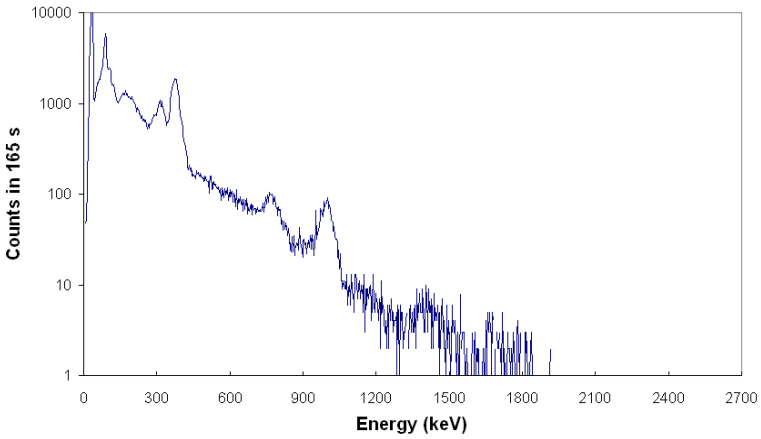

Figure 5. Scintillator spectrum of depleted uranium and ^{133}Ba.

3. Results

3.1 Single nuclides

For the single nuclide cases, both RIDs were able to correctly identify the nuclide, however the Detective determined the result in less time than the scintillator-based system.

3.2 DU and DU with other nuclides

The HPGe RID was able to identify DU as "depleted uranium" both alone and with other nuclides in less than 1 minute, and in some cases less than a few seconds. The time to identify increases as the number of gamma-ray lines in the spectrum increases, but remains at or near the 60 second requirement except for the DU and mixed nuclide standard. The DU plus mixed 1888 standard is identified in about 100 seconds as DU plus the 4 nuclides shown in Table 1.

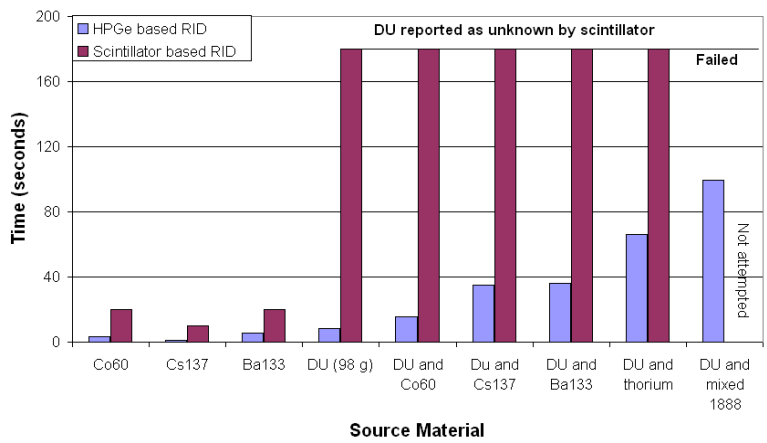

Figure 6. Comparison of time to identify single nuclides alone and with DU by HPGe and Scintillator RIDs.

The scintillator RID reported "unknown" in all cases where DU was the sample. For the mixtures with the ^{60}Co ^{133}Ba and natural thorium, the addin nuclide was reported with no DU reported. For the mixture of ^{137}Cs and DU, the scintillator RID only reported ^{137}Cs.

These results are shown in Fig. 6.

3.3 Other nuclides and combinations

Several other combinations of nuclides were tested. The results are shown in Fig. 7. The HPGe RID is able to identify the material in less than 130 seconds in all cases and generally less than 40 seconds. The background test was done in a room with concrete floors, walls and ceiling and the expected result was ^{40}K. The background dose is about 5 : Sv/h.

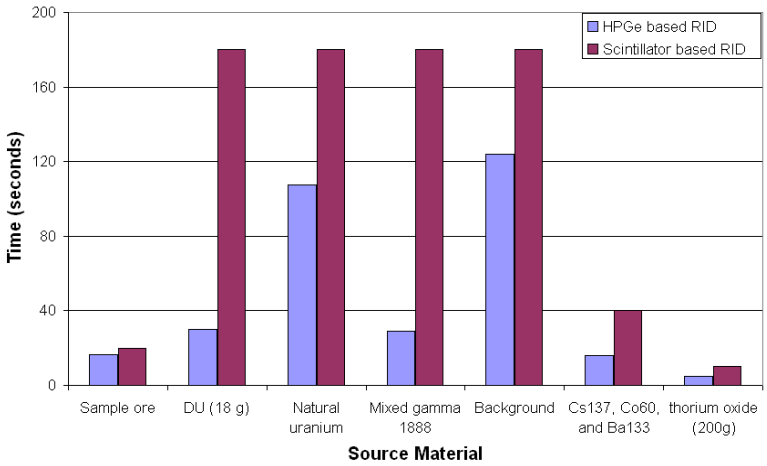

Figure 7. Comparison of time to identify multiple nuclide mixtures, uranium, and natural material by HPGe and Scintillator RIDs.

The scintillator RID reported “unknown” for the small DU sample and for the natural uranium sample. For the mixed gamma, only ^{241}Am and ^{60}Co were reported with ^{109}Cd and ^{137}Cs missing. For the background case, no nuclides were reported. For the thorium ore, any of the possible components (e.g., ^{232}Th, ^{228}Th, or ^{226}Ra) were accepted as the correct result, since the different requirements can dictate which result is displayed.

4. Conclusions

The results show that the HPGe-based RID can correctly analyze single nuclide samples and mixtures of materials in less time than a RID based on scintillation detectors. More importantly, the depleted uranium sample was correctly identified by the HPGe and not identified as DU by the scintillator, both in mixtures and even when DU was the only material present. This is possible because of the greatly improved resolution of HPGe compared to scintillators, which greatly improves the signal-to-noise ratio. Further work will concentrate on more complex mixtures.

4. References

1 Keyser, R. M., Twomey, T. R., and Upp, D. L., *An Improved Handheld Radioisotope Identifier (RID) for both Locating and Identifying Radioactive Materials*, HPS Midyear Meeting, January 2005

2 Keyser, R. M., Twomey, T. R., and Upp, D. L., *Performance of Light-Weight, Battery-Operated, High Purity Germanium Detectors for Field Use*, INMM Annual Meeting, July 2003

3 Twomey, T. R. and Keyser, R. M., *Hand-Held Radio Isotope Identifiers for Detection and Identification of Illicit Nuclear Materials Trafficking: Pushing the Performance Envelope*, World Customs Organization Conference, Sept. 2004

4 Keyser, R. M. and Twomey, T. R., *Performance of a Portable HPGe-Based Nuclide Identifier on Multiple Nuclide Mixtures*, INMM Annual Meeting, July 2004

5 American National Standards Institute, *Performance Criteria for Hand-held Instruments for the Detection and Identification of Radionuclides*, ANSI N42.34, January, 2004

6 International Atomic Energy Agency, *Technical / Functional Specifications for Border Radiation Monitoring Equipment* IAEA-TECDOC-xxxx, Rev 20E final draft April 2005

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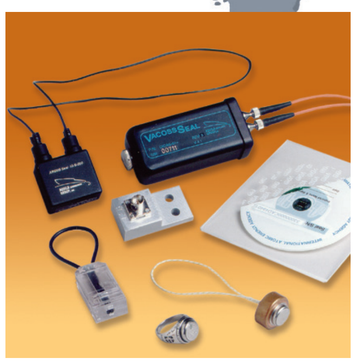

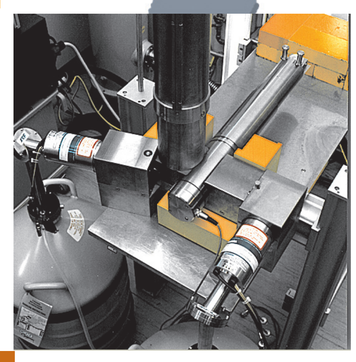

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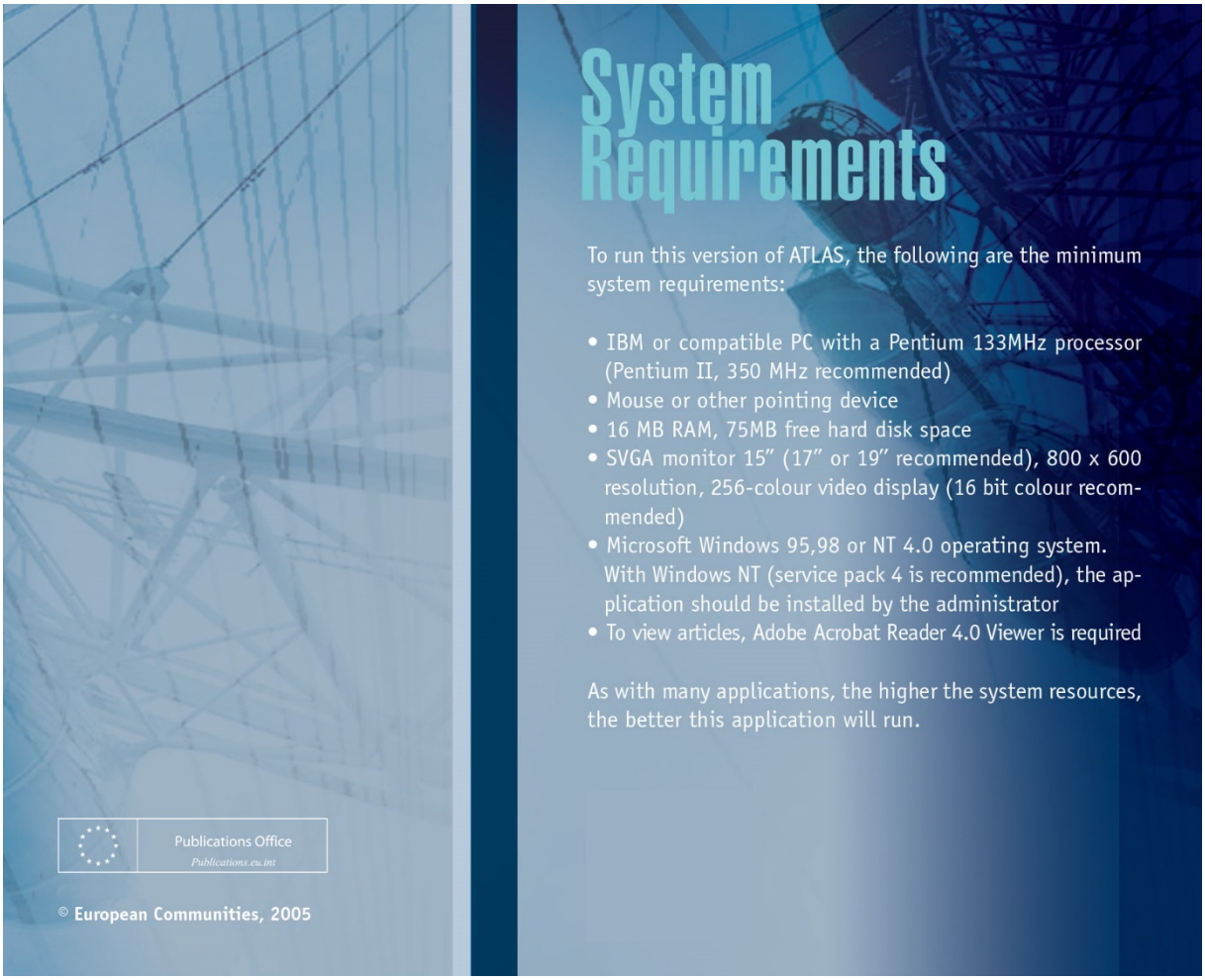

System Requirements

To run this version of ATLAS, the following are the minimum system requirements:

- IBM or compatible PC with a Pentium 133MHz processor (Pentium II, 350 MHz recommended)
- Mouse or other pointing device
- 16 MB RAM, 75MB free hard disk space
- SVGA monitor 15" (17" or 19" recommended), 800 x 600 resolution, 256-colour video display (16 bit colour recommended)
- Microsoft Windows 95,98 or NT 4.0 operating system. With Windows NT (service pack 4 is recommended), the application should be installed by the administrator
- To view articles, Adobe Acrobat Reader 4.0 Viewer is required

As with many applications, the higher the system resources, the better this application will run.