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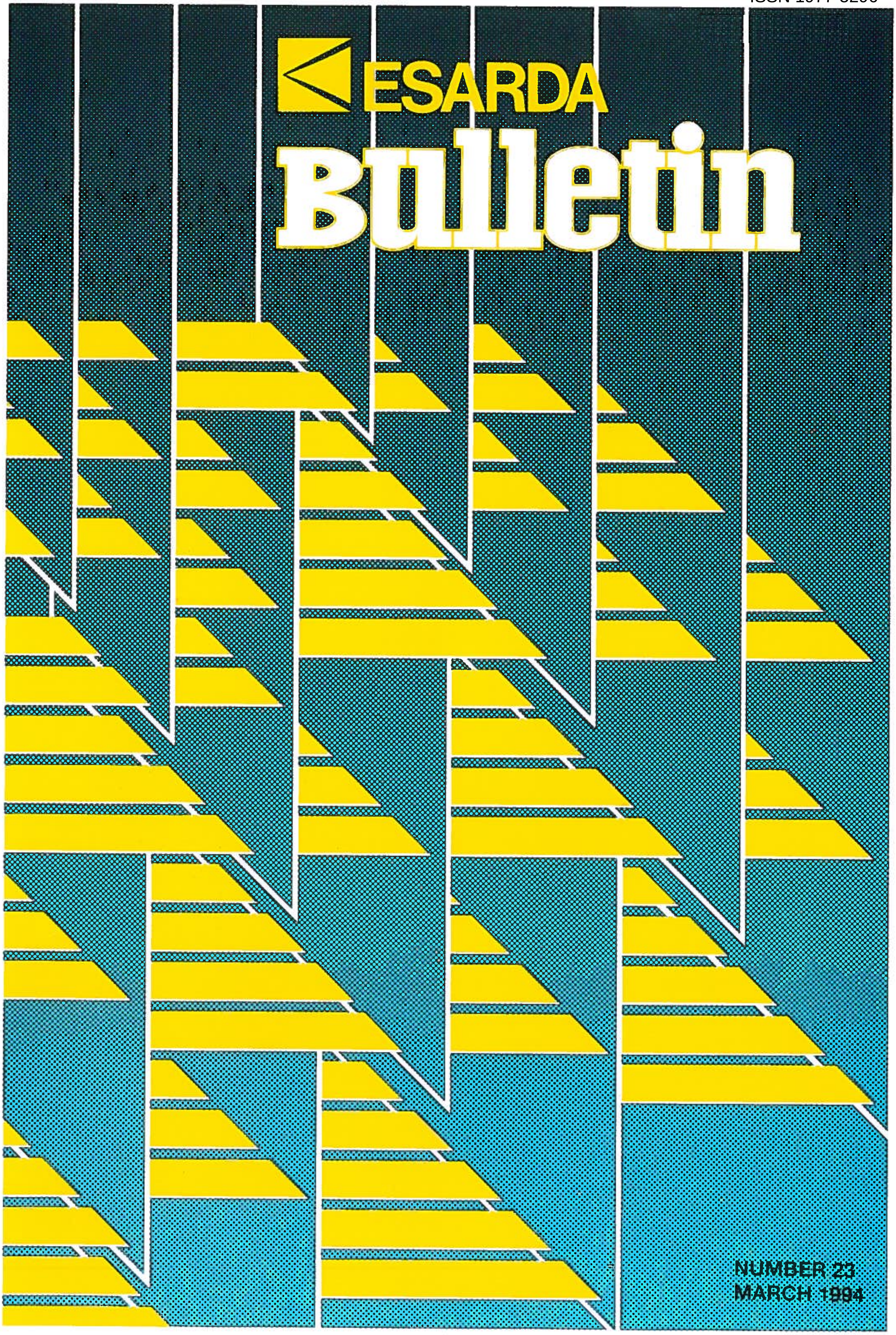


ESARDA Bulletin

NUMBER 23
MARCH 1994

The 16th ESARDA Meeting, which is of an internal nature, will be held from 17 to 19 May 1994 in the former Dominican Monastery "Het Pand", now property of the Ghent University and mainly used as a cultural centre. Het Pand has excellent rooms and facilities as needed by the Working Groups and for the ESARDA celebration and is also equipped with bar, foyer, tea-room and restaurant.

The ESARDA 16th Annual Meeting is a restricted meeting and the participation is limited to people from ESARDA member organizations and a few invited specialists.

This meeting will be based on individual and joint meetings of the ESARDA Working Groups, celebration of the 25th Anniversary of ESARDA, programming of the Working Groups' operations and prospective analysis of the ESARDA activities.

The registration fees per participant are 14,000 Belgian francs. The fees include:

- participation
- coffee and tea during conference breaks
- welcome buffet and a musical performance by the European choir (Carmina Burana by Orff) on Tuesday 17 May evening
- participation of accompanying persons in the social events
- conference folder
- secretariat services
- celebration of the 25th ESARDA Anniversary on 19 may followed by a reception

The meeting language is English. No translations are provided.

A visit to historical monuments and the city of Ghent and to the surroundings is planned.

For information please fax to Mrs. Anne VERLEDENS, Public Relations Officer, SCK/CEN Mol, Fax No. +32-14-332584.

Preview for Ghent

16th Annual ESARDA Meeting
Restricted participation

R. Carchon, P. De Regge
SCK/CEN, Mol

Ghent seems to be founded in the 7th century during a christianization effort and developed around two abbeys: the one of Saint Peter, founded by St. Amand, the other one of Saint Bavo, close to the river Leie.

Ghent has been built on the many islands at the meeting of the rivers Scheldt and Leie, which explains the many waterways that can be found in town.

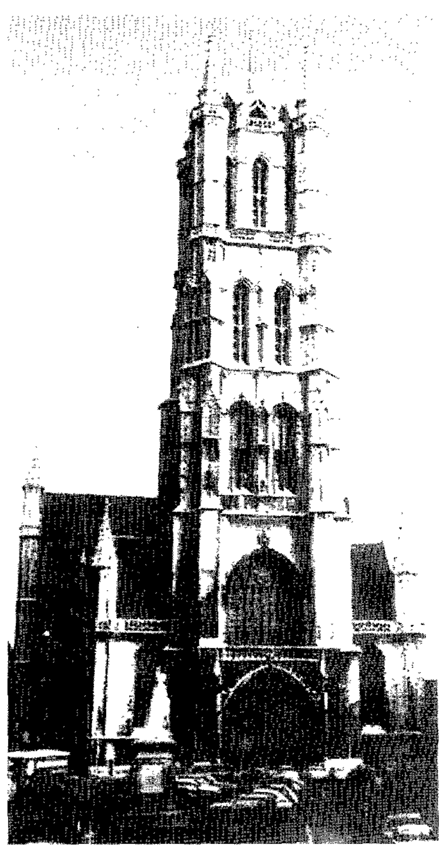
At the end of the 12th century, textile and cloth industry knew a serious expansion.

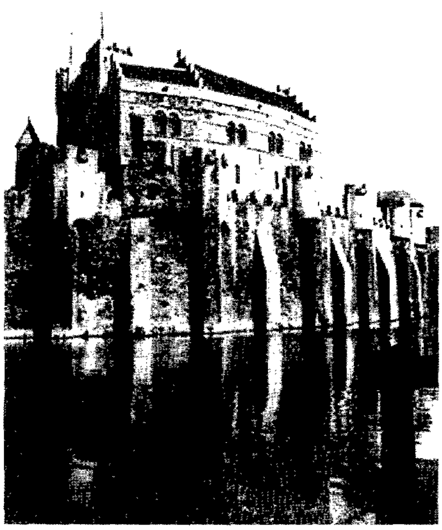
There have been many wars and conflicts, internal, among the different guilds, and external, resulting in an almost continuous changing of the political rulers.

In the 14th century, during the "hundred years war", Ghent, as a part of Flanders, in order to protect its industry, expressed sympathy for England against France.

In the 15th century, Flanders and Ghent became part of the reign of the dukes of Burgundy, with a high level of cultural development.

At the end of the century, the textile industry reached its end.





In the 16th century, there was a conflict with emperor Charles V concerning taxes.

At the end of this century, Ghent was perturbed by religious wars going on between dutch Calvinism and Christianity imposed from Spain.

Later, Ghent came successively under the influence fof Austria, under the reign of empress Maria-Theresia, of France, under Napoleon, and of the Netherlands until the independence of Belgium in 1830.

Ghent is the native city of emperor Charles V (1500-1558) who ruled over Western-Europe, from what is now Germany and Austria to Spain, and who belonged to the Habsburg family.

Apart from being a university city and the second port of Belgium, Ghent is now an important centre of industry. The city is rich in history and monuments, and offers a lot of touristic attractiveness.

Ghent counted among its inhabitants Maurice Maeterlinck, the francophone writer-poet and Nobel laureate.

Looking forward to meeting you soon in Ghent!

News about ESARDA

ESARDA Honorary Members

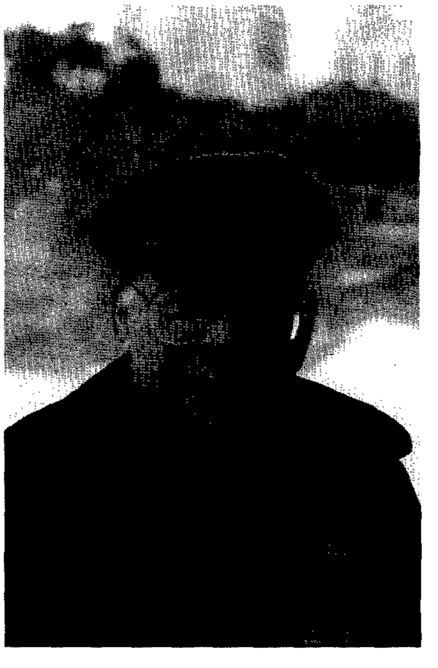
The ESARDA Steering Committee has decided to institute the position of ESARDA Honorary Member for people who have particularly worked towards the development of the association and for the achievement of its goals.

So far two members have been appointed:

- Sergio FINZI, who worked at the Commission of the European Communities, DGXII, Brussels
- Luciano STANCHI, who has retired from the Commission of the European Communities, JRC, Ispra, but is still editor of the ESARDA Bulletin

In Memoriam

With sadness we have to report that Dr. Reinhard Kroebel, the German member of the ESARDA Steering Committee, passed away on 23 September 1993 after long and heavy disease.



Dr. Kroebel started his professional career, which was entirely devoted to the nuclear fuel cycle, in 1965 at the EUROCHEMIC plant in Belgium working in the area of process control. After shutdown of this plant he moved to Bayer Leverkusen where he performed R&D in the area of the production of uranium(IV)nitrate. In 1973 he joined the Nuclear Research Centre Karlsruhe and built up the German Project for Reprocessing and Waste Treatment, the basic task of which was to carry out R&D for the first commercial German reprocessing plant to be constructed in Wackersdorf. He was the head of this project up to 1991 when it was cancelled as a consequence of a political decision.

In the nuclear community Dr. Kroebel was known as a highly competent scientist. He collaborated in several national and international advisory bodies dealing with various aspects of the peaceful uses of nuclear energy. For many years he engaged himself in the area of nuclear safeguards, being fully aware that non-proliferation issues are important factors regarding the public acceptance of the peaceful use of nuclear energy.

Because of his friendly and calm personality he was popular with colleagues within and outside his country. Even after the cancellation of the Reprocessing Project, which was especially hard for him who had invested his full energy and life for many years for its management, he never lost his optimism and fully engaged himself in new tasks until a cruel destiny prevented him from going on.

We will never forget him and honour the memory of him.

17th ESARDA Symposium

See the Preliminary Call for Paper on the third page of cover.

Who's Who in ESARDA?

(as of 15th March 1994)

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ESARDA Working Group Convenors

Techniques and Standards for Non-Destructive Analysis (NDA)
S. Guardini, CEC, JRC Ispra, Italy

Techniques and Standards for Destructive Analysis (DA)
P. De Bièvre, CEC, JRC Geel, Belgium

Reprocessing Input Verification (RIV)
vacant

Containment and Surveillance (C/S)
B. Richter, KFA Jülich, Germany

Low-Enriched Uranium Conversion and Fuel Fabrication Plants (LEU)
P.P.A. Boermans, FBFC, Belgium

Mixed Oxide Fuel Fabrication Plants (MOX)
R.F. Ingels, Belgonucléaire, Belgium

ESARDA Bulletin Editor

L. Stanchi, Italy

PERLA International Workshop on Calorimetry

Joint Research Centre, Ispra, Italy, PERLA Laboratory, 23-27 March 1992

This summary is reprinted from the proceeding of the Workshop

Abstract

The Joint Research Centre of the Commission of the European Communities and The EG&G MOUND Applied Technologies, have organized an International Workshop on Calorimetry as an Accountancy and Verification Tool for Plutonium and Tritium.

Developers and users of calorimetric assay instruments and measurements together with interested persons from international and domestic inspectorates and nuclear facilities have conducted a one week workshop on the calorimetric assay of plutonium and tritium.

The purpose of the workshop was to review the current status of calorimetric assay and make recommendations for further development efforts.

The workshop included presentations, planned demonstrations, measurements by participants, and discussion. Basic technology topics included the specific powers of radionuclides, the different types of calorimeters available for power measurements and the chemical and non-destructive gamma-ray measurements of plutonium isotopic composition. Methods to reduce measurement times have been discussed and demonstrated.

Instrumentation assembled for the workshop included a water bath calorimeter, dry transportable calorimeters built in the USA and dry transportable calorimeters built in England, along with gamma-ray systems for measuring plutonium isotopic composition and calculating plutonium effective specific power. Certified ^{238}Pu heat standards and well characterized bulk PuO_2 samples are available at the PERLA Laboratory of the Joint Research Centre.

These Proceedings constist of the extended summaries presented by the laboratories having participated to the workshop, the results of the measurements carried on and conclusions drawn.

Executive summary

This workshop was sponsored by the Joint Research Centre (JRC) of the Commission of the European Communities (CEC) and the United States (US) Department of Energy, Office of Safeguards and Security (DOE/OSS) as a continuation of the EURATOM/DOE Agreement for Cooperation in Safeguards Research and Development. The workshop was organized by the JRC, Institute for Safety Technology, Ispra, Italy and by EG&G MOUND Applied Technologies, USA. The workshop was convened to evaluate the current status of the calorimetric assay of plutonium and tritium and to make recommendations for further development and implementation.

The calorimetric assay of plutonium and tritium is used for accountability measurements by every facility having significant amounts of these materials. Calorimetric assay is also used for safeguards verification measurements during inspections by US DOE field offices.

The experience is significantly different in Europe and with the International Atomic Energy Agency (IAEA). Calorimetric assay had only limited use for the measurement of Pu contaminated waste in France and for safeguards Pu verification measurements by inspectors in the United Kingdom (UK). EURATOM and IAEA do not currently use calorimetric assay in inspections.

There are efforts underway, however, to use calorimetric assay routinely for Pu accountability measurements in the UK and for tritium accountability measurements at the JRC. This workshop is especially timely to provide recommendations to nuclear materials control and safeguard authorities and to instrument developers for the effective implementation of this technology now emerging in Europe. It is hoped that the results of this workshop will also be useful to the IAEA and EURATOM.

The workshop was structured to provide papers concerning all aspects of the calorimetric assay of plutonium and tritium and laboratory demonstrations of a variety of calorimeters as well as demonstrations of gamma-ray Pu isotopic measurement systems. The atmosphere was informal with discussions occurring freely during all parts of the workshop.

Papers were presented by users of calorimetric assay from Europe and especially from the US, by potential users from Europe and from both European and US developers in order to get a broad spectrum of viewpoints incorporated into the workshop evaluations and recommendations.

Five calorimeter systems from Europe and the US were included in the laboratory demostrations as well as three gamma-ray Pu isotopic systems utilizing the MGA Pu isotopic analysis code developed in the US and tested at the JRC. ^{238}Pu heat standards certified by EG&G MOUND were available as well as the PERLA PuO_2 standards.

From the papers, the laboratory demonstrations and from the continuing discussions, the workshop developed an evaluation of the current status of calorimetric assay as well as recommendations for further development and for effective implementation. Briefly, the evaluations and recommendations were:

- Calorimetric assay of Pu is more accurate and precise than neutron correlation methods but has a longer measurement time.
- Calorimetric assay of Pu and tritium should be used by plant operators for accountability measurements.
- The calorimetric, passive neutron correlation counting (PNCC) and gamma-ray Pu isotopic measurements should be used as a safeguards verification measurements system.
- An international measurement infrastructure including training of plant operators and inspectors in NDA technology, certified standards, and exchange programs should be developed.

The workshop participants further recommended that another workshop be convened at a later date, perhaps annually, to follow-up on the implementation of calorimetric assay throughout the international nuclear community.

ESARDA International Workshop on Passive Neutron Coincidence Counting

Joint Research Centre, Ispra, Italy, PERLA Laboratory, 20-23 April 1993

This summary is reprinted from the proceeding of the Workshop

Abstact

The ESARDA Working Group on Techniques and Standards for Non Destructive Analysis (NDA) has organized at the PERLA Laboratory of the Commission of the European Communities an International Workshop on Passive Neutron Counting as an Accountancy and Verification tool for Plutonium-bearing materials.

The workshop has reviewed the current status of passive neutron assay and made recommendations for further development efforts.

Developers and users of passive neutron assay instruments together with national and international inspectorates have conducted a three day workshop on the passive neutron assay of Plutonium.

The workshop consisted of presentations, demonstrations, measurements by participants and discussion periods.

Basic Technology topics included radionuclides nuclear data, Shift Register based instruments as well as Multiplicity Counters, HRGS measurements, uncertainty propagation models and performance evaluations of various instruments and techniques. Particular reference has been given to ESARDA NDA Performance Values and to IAEA International Target Values.

The measurement sessions have been performed on the well characterized PuO_2 and MOX PERLA standards.

The workshop has focused mainly on discussion sessions which provided the opportunity for the participants to evaluate the current status of performances of passive neutron assay and to develop conclusions and recommendations which have been included in these Workshop Proceedings.

Workshop evaluation of the status of passive neutron coincidence counting

A The conventional shift register technique

1. The two-parameter shift register technique has been used for a number of years and is satisfactory for many applications. It will remain the principal passive neutron measurement method for most material. The number of unattended applications will increase. Containment and surveillance methods can also be combined with NDA to produce integrated systems. More use of operator's passive neutron equipment by the inspectorates would lead to cost savings, if suitable authentication measures can be applied.
2. The performance of the method is described in a number of publications for a range of detectors and material types. The typical uncertainties for pure material are of the order of 1-2% without including the uncertainties on the isotopic composition values.
3. For the interpretation of all neutron techniques, it is essential that the isotopic composition is known. At very least the measured ^{240}Pu eff must be converted to Pu total.
4. HRGS (most often with MGA) plays an important role in the verification of the declared isotopic composition and has been extremely valuable. However if HRGS values are used for the analysis of the neutron results there is a significant bias in the case of high burn-up material caused by the error in the ^{242}Pu values which are obtained from the existing correlation. Other correlations, including the proposal described in /1/, should be tested with available data.
5. In order to determine the capability of the technique, performance values are needed. The definition of these values should be established. The Working Group appreciated the work carried out by the IAEA in using ANOVA to provide estimated Performance Values based on Inspector-Operator differencies. The Working Group took note of the fact that the error models of /2/ could lead to enhanced understanding of error performance and suggested that this approach should also be explored. Once this methods is proven with available data, the errors derived from historical performance can be compared with propagated error values. An algorithm for propagating the isotopic counting statistical and other errors has been developed and can be applied for such evaluations /3,4/.
6. The measurement of small samples using the two parameter shift register technique is currently being improved. This is being done by improving the design of the detector, careful background monitoring and reproducible sample handling.
7. The analysis of neutron concidence measurements should use an agreed set of nuclear data for the calculation of ^{240}Pu effective and alpha wherever possible. Currently used constants are given in annex.
8. In order to use technique to best advantage, the results from all evaluation techniques should be examined, i.e. the results from all known alpha, known M and uncorrected Reals should be consistent. (In some circumstances not all methods apply).
9. In the "known M" technique, the value of the multiplication is taken from empirical correlations between multiplication and effective fissile content. The method has been shown to work with assemblies and rods. Further work is required to validate the method for powders, pellets and MOX materials.
10. The conventional shift register technique has limitation when it is used to measure in-homogeneous and/or impure materials, particularly scrap and waste. For these materials alternative techniques are necessary.

B. Multiplicity analysis

11. The multiplicity analysis method gives more information, and can be used to determine the plutonium mass when alpha or efficency are unknown (in addition to the unknown multiplication).
12. Several types of electronics have been developed which essentially generate a histogram of multiplicity values. Several manufacturers are beginnig to supply electronic units.
13. The analysis is commonly based on the equations of /5/. The values of nuclear data currently used are given in Annex 1. Two semi-empirical corrections are required /6/. One is concerned with deadtime correction

and the other is due to the effect of multiplication varying within the sample. An analytical solution for deadtime /7/ has been produced and should be tested on available data.

14. This multiplicity analysis methods has reached a significant stage of development, and is used in the field. The uncertainty of the technique was estimated to be 3-5% for impure scrap and heterogeneous material (with uniform isotopic composition) over a wide mass range.
15. Another application of the technique is for waste (when the multiplication is small). Measurement of MOX waste drums with unknown alpha (around 1.0) and efficency gave an estimated uncertainty of 5-10% with a measurement time of about 1 hour.
16. An alternative analysis technique which may have advantages in some circumstances, is to calculate the multiplicity distribution to be expected from known sample properties (verification).
17. Recording the raw pulse train information provides a good opportunity for testing alternative data processing methods.
18. In order to test and compare different data acquisition and analysis systems, it is recommended that benchmark experiments be carried out. This would allow the measured values to be compared between systems and would also provide data which could be used to investigate alternative analysis methods.
19. To test the capability of the systems fully, characterized samples with a range of high alpha values are essential.
20. Because of the need for high, uniform efficiency, and low dead time, existing multiplicity detectors are expensive. However it is expected that the price could be reduced significantly by accepting a relatively small reduction of performance.

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/2/ M. FRANKLIN - "Statistical Models of MDA mesurements", Technical Mote Mo I.91.160 JRC Ispra, December 1991.

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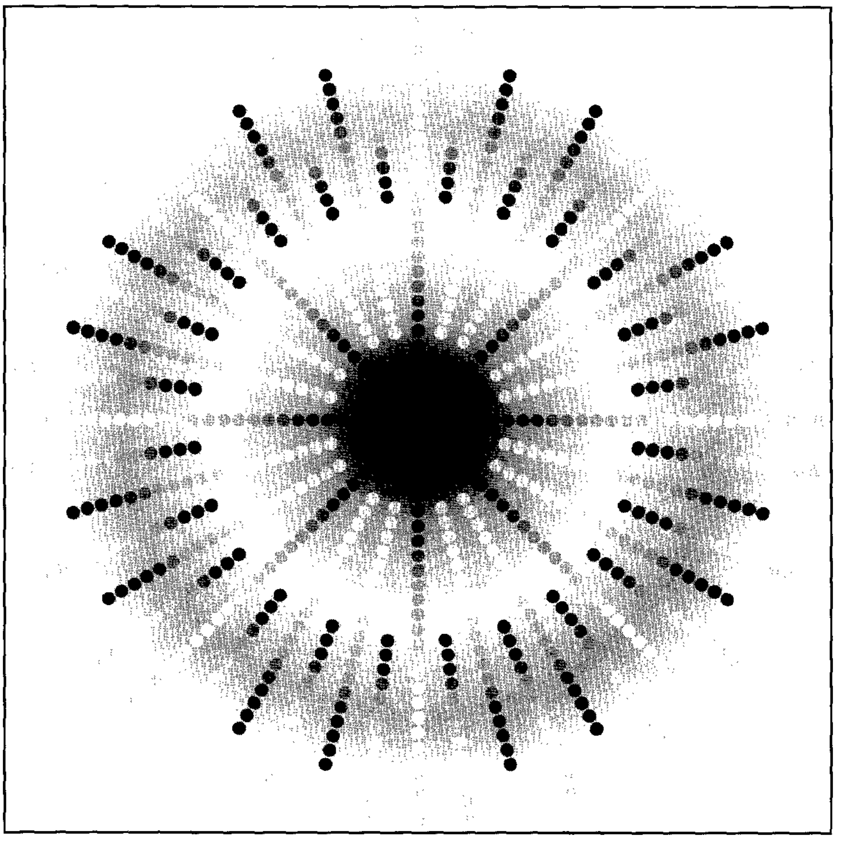
/4/ M.S.LU - Nuclear Technology, Vol. 102, No. 2, H 196209, May 1993. M.S.LU - Nuclear Instruments and Methods Phys. Res. A, A327, 544-550.

/5/ W. HAGE, D.M. CIFARELLI - Correlation analysis with neutron count distributions in randomly or signal trigerred time intervals for assay of special fissile materials, Nucl. Sci. and Eng. 89, 159-176 (1985).

/6/ M.S. KRICK, W.C. HARKER - User Manual for MULTI.

/7/ W. HAGE, D.M. CIFARELLI - Correlation analysis with neutron count distribution for a paralyzing deadtime counter for assay of spontaneous fissioning material, Nucl. Sci. and Eng. 112, 136-158 (1992).

The individual contributions to the workshop should be consulted for more details.



The role of non-quantifiable aspects in nuclear safeguards

R. Schenkel, W. Kloeckner
Commission of the European Communities
Directorate of EURATOM Safeguards

This paper is reprinted with the permission of INMM from the Proceedings of the 13th INMM Annual Meeting, Scottsdale, Arizona, USA, July 18-21, 1993.

Abstract

Nuclear material accountancy is a fundamental and unchallenged pillar of the EURATOM and IAEA safeguards systems. Nuclear material accountancy measures have also traditionally been used to derive quantifiable safeguards conclusions.

While the accomplishment of certain quantifiable conclusions is unquestionably an advantageous feature of nuclear material accountancy, it should not be overlooked, that the achievement of the safeguards objectives depends also to an increasingly large degree on so-called non-quantifiable aspects and their appropriate evaluation and follow up.

Since there appears to be almost no literature available on the subject, the aim of this paper is threefold; firstly to define and to present examples of non-quantifiable aspects and their use in safeguards, both at the facility and at the headquarters level. Secondly to discuss and analyse such aspects in view of their usefulness and conclusiveness for safeguards purposes. As illustrated in the history of safeguards, the way in which such non-quantifiable information is treated, judged and followed up both inside and outside the safeguards organisations, is of fundamental importance for the detection of diversions.

Third and last, ways are considered how such non-quantified aspects, wherever identified to be useful, could be better integrated in the implementation and evaluation of safeguards activities.

1. Introduction

During the last 10-15 twenty years, substantial changes have taken place in nuclear safeguards within the European Community, both in safeguards approaches and safeguards implementation.

One of the most significant and well-known change in the Euratom safeguards operations took place in the area of inspection verification technology.

The development, field test and implementation experience in this area has been described in numerous contributions of various Euratom staff members to safeguards conferences.

Euratom has always followed the concept of re-reverification of Basic Technical Characteristics (BTC) of installations as a fundamental part of its routine inspections. These activities permit the fulfilment of an obligation of the Euratom Treaty, namely to reverify, from time to time, the use of nuclear material and equipment and to check the continuing validity of the safeguards approaches applied.

Another change has occurred within the Euratom Safeguards Directorate itself. Due to the significant increase in the number and the complexity of some nuclear installations, a high degree of specialisation has taken place.

Whereas in the seventies "generalist inspectors" could cope with the majority of plants, it is now the "specialist inspector", who is needed for efficient and effective safeguards operations in the new generation of bulk handling and processing facilities. This specialisation includes not only the use of equipment, but also accountancy verifications, knowledge of plant and process verification techniques in complex plants as well as data evaluation and assessment aspects.

In safeguards discussions it occurs frequently that reference is made to non-quantifiable aspects. When challenged to explain in more detail what is meant and how such non-quantifiable aspects contribute to safeguards, people are often at a loss to provide explanations. Whereas nuclear material accountancy methods are well described and documented, it is therefore considered useful to have a closer look at what non-quantifiable aspects are and what they are worth in safeguards.

The purpose of this paper is therefore to review the performance of the Euratom safeguards system in the past with a view of identifying those components which have shown to contribute to the detection of diversions or important discrepancies. These components will then be distinguished whether they are quantifiable or non-quantifiable. It will become obvious from this evaluation that non-quantifiable aspects play a major role in present safeguards activities.

2. A few words about verification, indicators, detection, quantification and assurance

It is not the objective of this paper to enter into or create a theological discussion about the issues mentioned above, but it is nevertheless required to describe how the above terminology is used in this paper.

The mission of the Euratom Safeguards Inspectorate is "to **satisfy itself**, that in the territories of Member States

- ores, source materials and special fissile materials **are not diverted** from their **intended uses** as declared by the users
- **provisions** relating **to supply** and **particular obligations** from agreements by the Community with **third states** or an **international organisation** are **complied** with".

The basic objective of Euratom Safeguards is therefore to carry out, in line with the rights and obligations from chapter VII of the Treaty, the safeguards controls deemed necessary to achieve the "satisfaction" required by Article 77 of the Treaty. The main objective, however, is use verification.

For the achievement of this "satisfaction" or "assurance" and for the overall planning and evaluation of inspections, the Euratom Safeguards Directorate has established safeguards goals and inspection guidelines for each type of installation based on comprehensive diversion analysis.

Any significant discrepancy arising as a result of the evaluation of the safeguards activities is followed up thoroughly until it is resolved and thereby the assurance is obtained that materials could not have been diverted from their intended uses.

In this paper all indicators or results of activities, which can be expressed directly or indirectly (i.e. via calibrations) in amounts of nuclear material or number of items are considered quantifiable, the rest non-quantifiable.

In safeguards it is always very important to distinguish carefully between the operators and the inspectors viewpoint and a verification result on the one side and the safeguards assurance which can be obtained from this result on the other side.

For an inspector, for example, there are parameters in nuclear material accountancy, which cannot be directly quantified, sometimes even verified (examples are nuclear transformation or new measurement i.e. if used for MUF after a PIV, or accidental loss of nuclear material). There are, on the other hand, also in nuclear material accountancy, cases where the verification carried out by the inspectors may confirm precisely the declaration of the operator, yet the assurance provided by this verification alone may not be high (example: if an item is **only** verified bv weighing, there is no assurance that it actually consists of, or contains nuclear material).

Containment/surveillance or monitoring/logging systems on the other hand often provide results which are indirectly quantifiable (number of items moved from A to B, no underclared transfer through a penetration continuity of knowledge maintained, sensor signals ..)

The real problem however is, as with some individual nuclear material accountancy measures, of how **to quantify the assurance** which can be derived from a particular C/S arrangement or verification measurement in a given situation.

Both nuclear material accountancy methods and containment/surveillance and/or monitoring/logging systems contain quantifiable and non-quantifiable aspects. The "portion" of the non-quantifiable aspect depends on the amount of judgement of the inspector which is required to come to the acceptance or not of the individual check/activity performed. Examples for both cases will be given in chapters 3 and 4 of the paper.

The establishment of numerical figures for the safeguards assurance gained from a well chosen set of verification activities in achieving the safeguards goals is a subjective and extremely difficult, if not impossible task. This is not saying that numerical objectives, such as goals and statistical analysis in terms of detection and false alarm possibilities have no place in safeguards. They have in fact a very essential place in enabling procedures to be defined and resources to be deployed, taking into account the relative importance of the material involved and the accuracy of the instrument used.

With any safeguards measure, be it material accountancy, containment and surveillance, monitoring or other, the detection of diversion has as a first element the indication of a discrepancy or alarm and as a second element the subsequent follow up and assessment of the situation. It is then, of course, the thoroughness of the assessment procedure which controls detection and false alarm probabilities.

It is the objective of the next chapter to identify some of the above indicators of a discrepancy or alarm or diversion and to describe in what areas and by which activity these indicators have been recognised.

3. Indicators for a diversion, discrepancy or alarm derived from past safeguards performance

In banks or other high security areas, where the probability of a "diversion attempt" is high, the effectiveness of the security system can be assessed by reference to retrospective data. The effectiveness of such a security system for any future attempt can however change due to a significant change in the "motivation" for a "diversion attempt" and due to the unknown ingenuity and resources available to such a potential divertor.

A review of performance data from the past in order to identify indicators of diversions, alarms, or discrepancies therefore has limitations especially when, as in the Euratom safeguards areas, the diversions or diversion attempts have been very specific and rare (see examples under 3.1). Nevertheless, there are routinely alarms and discrepancies detected routinely and reported by inspectors in the field and during evaluations performed at headquarters. All those discrepancies/alarms could have been indicators for a potential diversion, and are followed up, assessed and - usually - resolved.

It was therefore considered useful to evaluate not only the few diversions or diversion attempts that occurred but also the important alarms/discrepancies reported during the last 3 years of Euratom operation in order to identify the type of discrepancies/alarms and the way inspectors have noticed the origin of the problems.

This survey covered in total more than 6000 inspection reports. Important discrepancies are reported at an average frequency of about 30-40 per year.

It should be noted that during the time period covered, the Commission imposed two sanctions in accordance with Art. 83 of the Treaty.

3.1. Indicators for detection of diversions or diversion attempts

During its 35 years of operation, the Euratom Safeguards Directorate detected three diversions/diversion attempts.

The first case concerned a diversion of about 200 t of natural uranium at the end of 1968 during a planned intra-community transfer. This case was detected by routine transit Euratom control, whereby the expected receiver did not confirm the receipt.

The second case concerned a diversion attempt in 1969 involving about 170 t of natural uranium again during a scheduled intra-community transfer.

The detection was triggered by a discussion between the inspector and the operator of the shipping plant, when the inspector raised doubts about the reliability of the envisaged destination. The transfer was subsequently stopped.

The third case of a diversion took place during 1984, involving about 50 t of depleted uranium. The diversion was detected by Euratom during follow up actions related to a non-corfirmed advance notification.

It should be noted, that the diversions involved an illegal transfer of the nuclear material concerned outside the European Community.

3.2. Indicators for detection of discrepancies/alarms from past experience

The evaluation presented is based on all important discrepancies reported in the last 3 years either by inspectors based on field operations or based on headquarters treatment and evaluation of safeguards relevant data. "Important" discrepancies are those involving either relatively large material quantities or problems of a generic nature influencing completeness, correctness and reliability of the operators nuclear material accountancy system.

The table below presents the results of the discrepancies reported from field operations, classified by type of discrepancy and proportion.

The individual discrepancies were analysed, as to the activity or indicator which triggered the detection of the problem.

The indicators/activities identified in this way and their frequency are given in table 2.

Table 1: Type and frequency of discrepancies reported from field operations

Type of discrepancy	Proportion (%)
1. Major shortcomings in operators nuclear material accountancy system (organisation, quality, completeness, correctness)	24
2. Major shortcomings in operators PIT procedures	20
3. Detection of undeclared material	11
4. "Material unaccounted for" not acceptable	10
5. Nuclear material not accessible for verification	7
6. Nuclear material/equipment not used as declared	6
7. Important discrepancy between declared and measured value	6
8. Loss of continuity of knowledge (C/S) not due to inspectors equipment failure	6
9. Discrepancy related to safeguards obligations	6
10. Detection of undeclared movements by optical surveillance/monitoring	2
11. Discrepancy between declared Basic Technical Characteristics and plant situation	2

Table 2: Indicators/activities which lead to detection of discrepancy during inspections

Type of indicator/Activity	Proportion (%)
1. Routine accountancy checks	30
2. Physical verification by item counting and identification	15
3. Inspector questioning validity of either accountancy, operating or source data	13
4. Inspector performing non-routine checks, or visits/checks in areas not normally inspected	8
5. Check of Basic Technical Characteristics, use of nuclear material, obligation	7
6. Physical measurements (weighing, NDA, DA, etc.)	6
7. Review of C/S or monitoring/logging data	6
8. Inspectors noticing un-usual operations, manipulations, transfers. damages, etc	6
9. Specific information received fom operators/contractors staff	5
10. Specific information received in Luxembourg headquarters	4

4. Non-quantifiable aspects and the role in nuclear safeguards

It is important at the outset to recall the main requirements for a safeguards system which allows for both, quantifiable and non-quantifiable aspects to become fylly effective:

a) institutional framework which, inter alia, ensures that the inspector is independent from the operator and the state;
b) legal framework which ensures, amongst other things
   - global coverage and centrality
   - full reporting from operators/holders
   - unlimited access of inspectors to materials, persons and documents at any time.
   - sanctions in the case of non-compliance
c) necessary resources (human, operational and hardware).

4.1. The role of the inspetor - the human factor

Analysing the data on discrepancies in more detail as far as the inspectors who detected/reported the discrepancies are concerned, the following correlations were noted:

a) there is a correlation with the length of the service in the European Commission, e.g. more than 4-5 years in the service,
b) there is a correlation with the period of time in the same group or cluster,
c) there is a correlation with high specific experience and knowledge on the operation of the plant inspected,
d) there is a correlation between the discrepancies detected per country and the nationality of the inspector; i.e. inspectors originating from the same country,
e) there is, as could be expected, a correlation with the personality and/or qualifications of inspectors.

In further pursuing some of the features identified in the previous paragraph, the following additional elements were identified:

a) a high plant specific experience and know-how is not only a pre-requisite for good inspection performance, but also saves significantly the time of operators.
   The other consequence of a high level of technical/safeguards knowledge of the plant inspected is however that it gives the inspector sufficient self-confidence to act and ask questions if he observes unusual operations, source data, etc., and to follow up the issue until resolved.
b) It follows from the above, for management purposes, to take into account the above features and not have rotation/mobility schemes which allow cycles to be too short and to ensure that the specific "plant safeguards history" is carried over, a factor very important, for example, for small facilities which are only inspected less frequently.
c) An interesting feature however was the role of "nationals" in detecting or identifying discrepancies in installations. The following elements were identified which may explain this fact:
   - language,
   - detailed knowledge of the mentality, culture and behavioural aspects which allow not only to identify more easily "weak points" or incomplete information, but also to acquire more quickly the confidence of people,
   - familiarity, especially when having worked previously in technical area in country of origin, with relevant regulations, procedures, licensing, organisational and other aspects,
   - independent status of Euratom inspectors as permanent Commission officials.

4.2. Other non-quantifiable aspects

The analysis in chapters 2 and 3 has shown, that non-quantifiable aspects play a major role in the following areas:

a) indicators arising from activities, where an operator's declaration is checked by the inspector and the detection of an alarm or discrepancy depends to a large degree on the judgement and professional experience of the individual inspector and/or the collective experience of the safeguards system as a whole;
b) indicators, which come from various sources outside the safeguards system including operators, Member State authorities, media, other Commission or Community institutions, third states or the IAEA

In the following, the non-quantifiable aspests of the indicators/activities given in table 2 are discussed in some more details:

a) **Routine accountancy checks**
   Most of the important discrepancies were detected by routine accountancy checks. It is often overlooked, however that same accountancy checks require a high degree of judgement by the inspector. Examples are inventory changes which cannot directly be verified, or which took place without inspector's presence (examples are nuclear transformation, accidental loss, MUF, measured discards, category changes, shipper receiver differences, new measurements, exemption, etc.). Even if the documents presented by the operator are consistent, the acceptance - or not - require a high degree of experience and judgement.
b) **Item counting and identification**
   The non-quantifiable aspect involved in this activity refers to the recognition of the item as the one declared and its identification. This includes, for example that inspectors would recognise whether the declared transport container, bird cage or containment arrangement was used, or a different one than usual thus triggering questions from the inspector.
   It should be noted for clarity however that most of the discrepancies referred to in table 1 were detected by routine counting and number or tag identification.
c) **Examination of validity/plausibility of operator's accountancy, operating or source data**
   This activity is obviously a part of the routine accountancy checks referred to under a) above. It was treated separately in the statistical evaluation, because this feature alone was often the reason for the detection of a discrepancy. This high "score" is very positive, as it reflects that inspectors do not compare in a mechanical fashion a source document with the entry in the accountancy ledger, especially, when both documents are purely "internal", i.e. if a "confirmation" document from a shipping operator is not available. Examples for

discrepancies detected under this category were related to certain material transfers, which had substantially higher values in the control period than usual for normal operations and where operator's source data and accountancy data were consistent.

d) **Inspector performing non-routine checks, or visits/checks in areas not normally inspected**
This activity, which turned out to be the "fourth-important" in table 2, has often been discussed under the term "unpredictability" but this should not be confounded with randomisation or unannounced inspections.
The cases mentioned under this heading include the following type of activities:
- examinations in plant areas or on materials which were inspected less frequently in the past
- examination of operational and source data in plant laboratories control process areas
- application of different verification techniques than those "normally" applied.

It means a certain degree of "unpredictability", of where and how and to what extent the inspector performs accountancy and physical verification.

It is required that this aspect be further analysed and promoted during for example, plant inspection history discussions, and as part of safeguards approaches.

e) **Check of Basic Technical Characteristics, use of nuclear material, obligation codes**
The examination and re-examination of Basic Technical Characteristics, the use of nuclear material/equipment and the control of safeguards obligations form an important and routine part of Euratom safeguards operations. The activities provide the inspectors with a substantial upgrading or the maintaining of their knowledge about the plant and its operation and provide the possibility of access to all parts of the plant and all persons involved. It can also be seen as a concept of complete transparency by the operators concerned which however requires a high degree of judgement by an experienced inspector.

f) **Review of C/S or monitoring/logging data**
With regard to the review of optical surveillance data no difference is made between in-field and headquarters activities. The introduction of automated front-end and back-end processing capability in this area will relieve the inspector from a very important tedious and tiring activity. The fact, however that the evaluation technique, is "pre-programmed and fixed", places the onus on the inspector to evaluate carefully in advance the possible diversion scenarios and concealment methods. It is, in our view however, necessary to perform in some cases a certain amount of additional "human" control on the data evaluated automatically.
As was stated previously, a certain amount of results obtained from C/S and monitoring/logging data are either directly (level, weight, NDA) or indirectly (item count from C/S) quantifiable. The major non-quantifiable activity in this area is the check of the integrity of the containment and the identity and integrity of seals during seals verification.

g) **Inspectors noticing unusual operations, manipulations transfers, damages etc.**
This is a feature directly related to the inspector's knowledge of the plant operation/procedures and his attention in carrying out his duties.

h) **Specific information received**
On some occasions inspectors noticed either discrepancies in statements made by different people in the plant or obtained indications of problem areas.
Apart from the indicators/activities derived from table 2, there are other non-quantifiable aspects which play an essential role in nuclear safeguards, either in field or during headquarters activities.

i) Transit control matching: this is by far the most important indicator of problems, not only, as shown, to detect diversions, but also, for example, to "detect" and identify new holders of nuclear material. An additional essential activity is the follow up of advance notifications.

j) A further significant aspect is the internal follow up and both the formal and informal discussion culture of inspection results together with other inspectors colleagues. It is then where observations, difficulties, uncertainties etc., i.e. "non-quantifiable" aspects are further discussed and assessed and that ideas which may lead to specific follow up investigations to resolve the issues concerned are developed. Additionally the recruitment and training of inspectors, the staff policy to achieve a high degree of motivation, competence and self-confidence, rotation schemes, questions such as the "national effect" need permanent attention and support

k) Another essential aspect of headquarters based evaluation and assessment is the regular information and discussion of actual nuclear policy, nuclear programmes and non proliferation related issues. This information is useful for the inspectors with regard to the safeguards of nuclear material and equipment, for example as related to imports and exports.

l) Resource deployment is unquestionably the most important non-quantifiable management responsibility.
In this area, the following features are relevant:
- recognition of, speed and thoroughness of follow up of any anomalies and discrepancies,
- flexibility in operations to respond to specific circumstances,
- taking into account the safeguards history of plants or countries or regions, the political system (democracy, free press, transparency of nuclear programmes).

m) Last but not least, the Euratom Safeguards Inspectorate is receiving regularly sareguards relevant information not only from the Commission or Community institutions, but also from Member States, operators third states and in particular from the IAEA which may require direct follow up.

5. Conclusions

The paper presented is a first attempt, based on an empirical approach, to identify with concrete examples non-quantifiable aspects in nuclear safeguards.

The approach chosen was based on the examination of the performance of the Euratom Safeguards Inspectorate with regard to diversion detection and the detection of alarms or discrepancies over a 3 years period.

The analysis yielded a variety of examples of non-quantifiable indicators or activities which play an important role in nuclear safeguards.

The major conclusions from the analysis/discussion can be summarised as follows:

- overall safeguards assurances are difficult, if not impossible to quantify,
- there are quantifiable and non-quantifiable aspects involved in both, nuclear material accountancy and in other safeguards measures such as C/S, monitoring and logging systems,
- there are essential pre-requisites necessary in order that non-quantifiable aspects can become effective,
- the inspector's competence, profile, and knowledge of the plant and the related safeguards concept is of the utmost importance,
- more flexibility and unpredictability in inspection activities is another important element,
- there are headquarters activities which are important for the effectiveness of a safeguards system and are non-quantificable by their nature.

The analysis carried out has left open a number of questions which will be further pursued.

Fork Detector Measurements on LRW Spent Fuel

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Abstract

For fuel assemblies that have an initial enrichment above 3,75%, it is essential before shipment to reprocessing or storage, to make an appropriate declaration of burnup and to confirm the calculation by a reliable experimental check, for spent fuel management reasons.

In the case of light water reactor (LWR) fuel, the Fork detector is a suitable measurement device for this purpose, as an estimation can be made of burnup and cooling time of the irradiated fuel.

The data provided by this device require some corrections to allow an appropriate estimation of burnup.

In this paper, an account is given of the use of such device on a routine basis at a nuclear power plant.

For this study, computer calculations with ORIGEN 2 have been performed, that simulate reality and which give a good estimate of the neutron and gamma emission rates from different isotopes.

1. Introduction

Spent fuel shipment to the reprocessing or storage plant is normally accompanied with the operator declared information, consisting of the following data:

- from operation:
  - initial ^{235}U enrichment (%)
  - irradiation history (days)
  - cooling time (days)
- by calculation:
  - burnup (MWd/tU)

Safety reasons (criticality) and current nuclear materials management at the reprocessing plant require an independent verfication of burnup of spent fuel as calculated by the operator. This is true especially in the case of high initial enrichments - e.g. above 3. 75% - in which case a correct estimate of the average burnup value in the upper and lower extreme parts (with a reduced burnup) is now requested.

In earlier experiments /1/, performed in a safeguards context, axial scans showed a reduced neutron and gamma emission rate at the outer ends of a fuel assembly, yielding in a reduced burnup value.

Several methods can be applied to determine the burnup of spent fuel and one of them is the Fork detector, a passive non-destructive assay (NDA) measurement device for the determination of neutron and gamma emission rates of spent fuel assemblies. It is a Los Alamos concept /2/ and the original device contained per detector arm 2 fission chambers (1 bare and 1 Cd wrapped) and 1 ionization chamber. This device was designed to determine for safeguards purposes, the burnup, plutonium content and cooling time of spent fuel assemblies and to estimate the boron content of the reactor pool. The device already proved its capabilities on several assemblies at the TIHANGE Nuclear Power Plant #1 /1/. It is possible to perform axial scans as well as mid point measurements.

An alternative device is now used for making independent estimates of the burnup of spent fuel assemblies for safety reasons before shipment to a reprocessing plant. For this purpose, a simplification was introduced by limiting the number of fission chambers to 1 per arm, avoiding any energy discrimination of the emitted neutrons. The boron concentration in the storage pool could be considered fairly constant at 2200 ppm ± 4%. In order to have the least sensitive signal to that boron fluctuation /3/ the cadmium wrapped fission chamber has been used.

A study has been made /4/ of the influence of the introduction of deviations on operator declared values such as irradiation time, cooling time, power and burnup on the value of burnup, deduced from the measured neutron count rate and gamma current, in order to determine the sensitivity of the method to possible erroneous data communication by the operator. This sensitivity study is crucial for the interpretation of these measurements.

In this paper, a description is given of the use of the Fork detector for the identified purposes. A technical description is given in paragraph 2. Paragraph 3 contains the basic correlations and underlying physics, while experimental results are given in paragraph 4, containing calibration effort, tests and operational results.

2. Equipment

2.1. Mechanical Part

The mechanical part is composed of a stainless steel box, with two arms that embrace the fuel assembly. Each arm of the device contains two sensors, one for neutron detection (fission chamber) and one for gamma detection (ionization chamber). They are embedded in high density polyethylene body which is contained in a 1 mm thick cadmium layer.

The device is installed on a vertically moving elevator for fresh fuel, having a height of 4 m and a cross-section of 21*21 cm² to accommodate a PWR fuel assembly.

A funnel was put on top of that box to facilitate the introduction of the irradiated fuel assembly and serves as a support for the NDA device.

The device remains fixed on the fresh fuel elevator.

A possibility exists to detach it and put it aside at every occasion that fresh fuel is brought into the storage pond.

For measurement campaigns, the box is put on the bottom of the storage pond, serving as a reference point, and the irradiated fuel is introduced by the usual tools.

The link between detector body and data acquisition unit is made by cables guided through a watertight flexible tube.

One extreme of that tube is fixed on the wall of the storage pond and allows cable connection with the electronics at the moment of the measurement campaign.

Under normal conditions, the device remains permanently installed in the storage pond of the reactor. Only the data acquisition unit is approached and connected at every campaign. The system is disconnected at the moment of fresh fuel introduction, and can be attached at the side wall at every such activity, and easily reassembled in a quick way. The spent fuel assembly remains connected to the hook by the handling bar.

2.2 Electronic part

The electronics chain contains all necessary components required to permit an adequate signal treatment:

- the neutron chain: high-voltage power supply, preamplifier, counter/timer,
- the gamma chain: high-voltage power supply, picoamperemeter.

Modular NIM units were used where possible.

All the electronic equipment, as well as the data handling system, is housed in a mobile 19" rack.

2.3 Data handling System

The system is steered by a personal computer (PC), that collects the data from the electronic part and the height measurement device and assures data treatment, but does not affect fuel movement.

2.4. Height measurement device

The height measurement device assures an exact indication of the position of the assembly in relation to the measurement device.

It is composed of a decoder and appropriate display, with RS-232 interface.

The display is fixed on the railing of the fuel-handling bridge, while the decoder is mounted on top of it.

The display gives this information to the operator of the tool, while RS-232 interface permits the transmission to the PC.

3. Basic correlations

The following abbreviations will be used throughout this text:

- n: neutron count rate
- G: gamma current
- B: burnup
- T: cooling time

The two basic correlations are:

$n = a\,B^b$ (1)

$G/B = c\,T^d$ (2)

wherein a, b, c and d are coefficients, obtained by calibration.

For neutron calibration purposes, a series of assemblies of the same enrichment is measured and a power function fitting is made of the neutron count rate versus operator declared (calculated) values of burnup.

The neutron count rate is proportional to the amount of ^{242}Cm and ^{244}Cm that has been produced by successive neutron capture by ^{238}U and the transuranium chain that results from this phenomenon. This means that the neutron count rate is proportional to the fluence seen by the assembly. On the other hand, the burnup of the assembly is proportional to the integral of fluence times the fissile mass of the assembly. This means that from two assemblies that have been submitted to the same fluence, the one with the highest initial enrichment obtains the highest burnup, or said otherwise, from two assemblies with the same burnup, the one with the highest initial enrichment will have the lowest neutron emission rate.

In order to refer all data to the same reference point that corresponds to end of irradiation, the neutron count rate n (measured) of correlation /1/ has to be corrected /5/:

$n = F1 \cdot F2 \cdot n$ (measured)

with $F1 = e^{\lambda_{244} T}$

$F2 = n_{244}/n$, calculated by ORIGEN 2

wherein:

λ_{244} : radioactive decay constant of ^{244}Cm

T : cooling time

n_{244} : neutron count rate of ^{244}Cm

n : total neutron count rate

No attention has been devoted to refer all assemblies to a reference enrichment; a calibration curve was made per initial enrichment.

Correction factor F1 is calculated on basis of the cooling time, either operator declared or measured, while F2 is calculated by ORIGEN 2.

ORIGEN 2 makes use of neutron half-lives and neutron yields from spontaneous fission and (α,n) reactions with an accuracy of 5 to 10%, and 15 to 30% resp./6/.

The neutron emission rate is calculated for the main heavy metal isotopes, after irradiation as well as after a certain cooling time. The main contributors to the heavy metal contents of LWR fuel after irradiation are: ^{238}U ^{238}Pu ^{239}Pu ^{240}Pu, ^{242}Pu, ^{241}Am, ^{242}Cm and ^{244}Cm.

Depending on burnup and cooling time, the most important contributions to the neutron emission rate come from ^{242}Cm ($t_{1/2}$=163 days) and ^{244}Cm ($t_{1/2}$=18.1 years) /7/. The first one decays faster, so its contribution is most important immediately after irradiation, but decreases by a factor of 1000 after 5 years of cooling.

The other isotopes of heavy metals have a more or less constant contribution, independent of cooling time, with the exception of ^{241}Am, which evolves by a factor of 10 over 5 years of cooling.

For specific burnup values greater than 20,000 MWd/tU and cooling times above 18 months, the principal neutron source is ^{244}Cm.

Some drawbacks have been mentioned in the mean time: the correction formalism for cooling time relies on the ^{244}Cm isotope part of the neutron emission rate. It has been mentioned /8/ that this correction formalism in particular is sensitive to the irradiation history, the latter being continuous or discontinuous, which means that irradiation cycles have been interrupted by some years.

This means that an ORIGEN 2 calculation would have to be made for every measurement, corresponding to the exact irradiation history. For the application described here, an approximation could be made, as most of the assemblies had a similar irradiation history and cooling time. A graphical representation is made in figure 1.

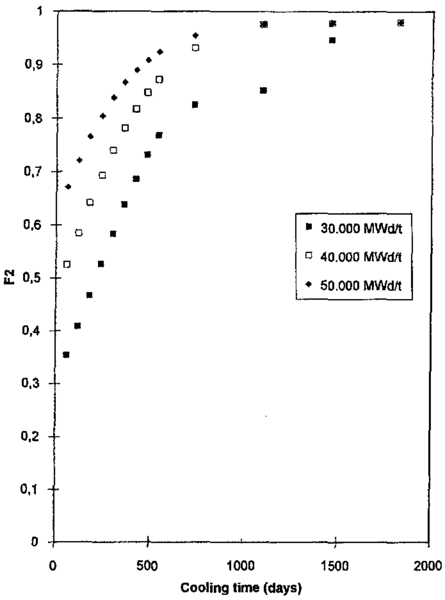

Figure 1 : Correction factor F2

From the gamma emission rate, a value of the cooling time can be deduced through correlation (2), but because of the lower accuracy (5%) in comparison with the neutron results, this measurement result is not used for the determination of the cooling time, but only as a check of the operator declared value.

The sequence that was applied in the software is as follows:

- estimate B from correlation (1) n measured and T calculated
- estimate B from correlation (2) G measured and T calculated
- estimate T from correlation (2) G measured and B measured via n

4. Experimental results

4.1. Calibration

The establishment of calibration curves requires neutron and gamma measurements on a series of spent fuel assemblies with a great diversity in burnup and cooling time.

The availability at the storage pond has put a limitation on the choice that could be made:

- Initial enrichment: we limited ourselves to an initial enrichment (IE) of 3.9%. Additional curves for other initial enrichments could easily be added. These could be obtained by simulation through a calculus followed by an experimental verification with assemblies of another initial enrichment.

- Burnup: a broad burnup scale, say from 10,000 to 50,000 MWd/tU would be ideal but was limited in practice. The operator declared burnup was a mean value over the entire lenght of the fuel assembly; the maximum value at mid assembly position was 6% higher than the mean value.
- Cooling time: cooling time is calculated as the difference between end of irradiation and measurement date. Only two different cooling times were available. This means that the calibration curve is an exact curve and not a statistical adjustment.

Five spent fuel assemblies have been used for establishing the calibration curve for 3.9% IE. They were available at the installation. The basic characteristics, together with the measurement results are given in table 1, giving the identification number (ass), calculated burnup (BU), initial enrichment (IE), the end of irradiation date, cooling time (CT), measured neutron count rate, correction factors F1 and F2, gamma current as measured by the ionisation chamber.

The choice is justified in the following way:

- A02, A09, A06: these assemblies give a complete scale of burnup values, available at the plant.
- A56, A54: both assemblies have been charged in symmetric positions during the same cycle. The aim of this measurement is a comparison of two quasi identical assemblies.

Calibration measurements consist in an axial scanning over the complete length of the assemblies.

Correction factor F1 is calculated by the software based on the declared cooling time. This cooling time also determines factor F2 that is taken from an analytic curve. The used approximation for F2 is that cooling time is minimal 13 months and burnup lies between 30,000 and 50,000 MWd/t.

Calibration curves are part of the software and are the key elements to data evaluation.

Curve fitting was made by the DEMING code.

Calibration curves using mid-assembly neutron count rates and gamma current are:

(1) $n=3.6716\ 10^{-4}\ B^{3.9480}$

(2) $G/B=925.85\ T^{-0.28657}$

with G in nA

B in GWd/tU

and the former is graphically displayed in figure 2. Figure 3 gives the axial scan of all the assemblies used for the calibration.

The mean burnup over the extreme of 50 cm was taken according to the neutron emission rate at 25 cm from the top.

Table 1: Selection of assemblies for calibration purposes

ass.	B (MWd/t)	IE (%)	end irr.	CT (d)	n/s	measurement (F1)	F2
A02	47895	3.9	01.09.91	464	2057.7	1.0499	.9
A09	36489	3.9	01.09.91	463	768.9	1.0498	.845
A06	32760	3.9	05.05.90	948	414.3	1.1045	.96
A54	45547	3.9	01.09.91	464	1767.6	1.0499	.9
A56	45495	3.9	01.09.91	464	1723.9	1.0499	.9

The exponent of approximately 4 is typical for LWR fuel.

4.2. Tests

Tests were made on some assemblies, in order to have an idea of the performance of the calibration effort.

Taking into account that the reactor is operated in a standard way, the choice in assemblies, serving for qualification purposes, is rather limited.

Available assemblies are given in table 2, comparing the calculated and experimental burnup and the deviation between both.

4.3. Experimental Results

The equipment has now been in use for more than one year, and all shipments are accompanied by a measurement report, confirming the operator declared value.

Other initial enrichment values will be added to the list of calibration curves.

An overview of the results is given in table 3, with a comparison of calculated mean burnup, derived maximum burnup (corresponding to mid-assembly position and being 6% higher than the mean value) and the experimental result.

Burnup values derived from these measurements confirm the calculations within 1%.

5. Conclusions

In the context of a sound nuclear materials management for criticality safety purposes, it is essential for fuel assemblies with high initial enrichment that a reliable check exists of the operator declared value of burnup. The extreme parts of a fuel assembly show a lower burnup than the central parts and show different criticality characteristics at the reprocessing plant /1/.

The Fork detector was chosen to make this independent check, and has now been in use for more than one year.

Calibration curves were obtained corresponding to correlations (1) and (2), limited to fuel assemblies with an initial enrichment of 3.98%; the set of calibration curves is now being extended to other enrichments.

The influence of the presence of Gd in the assemblies is negligible, taking into account that Gd is burned after one year of normal operation of the reactor.

Assemblies with a quasi identical burnup give a quasi identical neutron count rate.

The instrument is capable to measure all assemblies, independent of burnup, cooling time and initial enrichment. The inherent limits in the programme come from the correction factor formalism that is applied on the neutron count rate, and that is presented in this case by an analytic curve. These limitations can be changed according to the requirements.

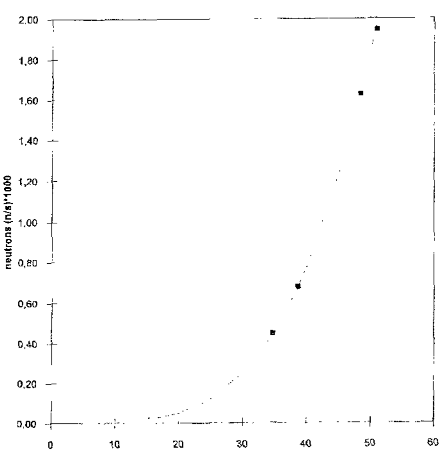

Figure 2: Calibration curve for 3.9% initial enrichment

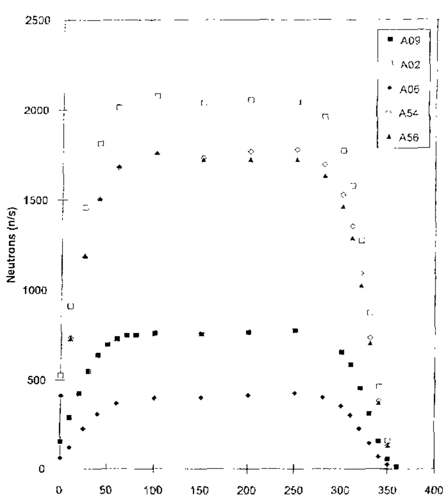

Figure 3: Axial scans of calibration assemblies

Table 2: Selection of assemblies for test purposes

ass.	Bmean (MWd/t)	IE (%)	end irr.	CT (d)	Bmax (MWd/t)	Bexp (MWd/t)	delta %
A05	32749	3.9	05.05.90	958	34713	34866	0.44
A11	36560	3.9	01.09.91	474	38753	38547	-0.53
A26	44624	3.9	01.09.91	474	47301	46135	-2.47
A03	47939	3.9	01.09.91	474	50815	49796	-2.01
A10	36527	3.9	01.09.91	473	38718	38683	-0.01
A36	44492	3.9	01.09.91	473	47161	45681	-3.14
A50	36584	3.9	01.09.91	507	38779	38916	0.35
A21	37949	3.9	01.09.91	507	40225	39652	-1.42

delta = relative difference between calculated maximum and experimental burnup.

Table 3: Experimental results

ass.	Burnup mean MWd/tU	Burnup max MWd/tU	Burnup exp MWd/tU	Burnup 25 cm MWd/tU	delta %
A09	36489	38678	38750	29333	+0.19
A10	36527	38718	38875	29637	+0.41
A11	36560	38753	38687	29599	-0.17
A12	36509	38699	38557	29442	-0.37
A13	37973	40251	39844	31884	-1.01
A14	38115	40401	40081	32123	-0.79
A15	38072	40356	40013	32260	-0.85
A16	38007	40287	39890	32343	-0.99
A17	38217	40510	40215	32634	-0.73
A18	38227	40520	40017	32768	-1.24
A19	38085	40370	39790	32094	-1.44
A20	38101	40387	40138	32750	-0.62
A21	37949	40225	40053	32291	-0.43
A22	38023	40304	39964	32181	-0.84
A23	38060	40343	40007	31900	-0.83
A24	38081	40365	39930	31856	-1.08
A25	44292	46949	46837	36990	-0.24
A26	44624	47301	46876	38139	-0.90
A27	44340	47000	46106	37638	-1.90
A28	44651	47330	46649	37757	-1.44
A29	44400	47063	46462	37370	-1.28
A30	44459	47126	46184	37438	-2.00
A31	44594	47269	46996	38174	-0.58
A32	44547	47219	46795	37825	-0.90
A33	43765	46390	46207	36651	-0.39
A34	44254	46909	46003	37725	-1.93
A35	44043	46685	46431	36672	-0.54
A36	44492	47161	46826	37148	-0.71
A37	44393	47056	46613	37410	-0.94
A38	44360	47021	46369	37728	-1.39
A39	44070	46714	46199	36127	-1.10
A40	44408	47072	46876	37469	-0.42
A45	45102	47808	47898	39691	+0.19
A49	36553	38746	38840	29893	+0.24
A50	36584	38779	39176	30391	+1.02
A51	36543	38735	38632	30004	-0.27

Current measurements confirm the calculated burnup values within 1%.

The measured values of the extremes of the assemblies serve to make an estimation of mean burnup of the last 50 cm of the assembly.

The data accumulated during the fuel evacuation campaigns will allow to refine the calibration curves by taking these data into account and by this way improving the statistics of the data.

Measurement are not influenced by neighbouring assemblies in the storage positions.

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1993 International Target Values for Uncertainty Components in Measurements of Amount of Nuclear Material for Safeguards Purposes

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Note from the editor. This paper is the result of continual efforts and world wide discussions regarding the establishment of reliable ITVs. It completes and updates two preceding papers: "1983 Target Values for Uncertainty Components in Fissile Element and Isotope Assay" , and "Random Uncertainties in Sampling and Element Assay of Nuclear Materials - Target Values 1988", published in the ESARDA Bulletin No. 6, April 1984 and ESARDA Bulletin No.13, October 1987 respectively.

Because the importance of the matter the present paper is published both here and in the JNMM.

The paper has been reviewed by the ESARDA Board. It appears suitable to publish the paper with the same title given before the tables instead of the original title "1993 International Target Values for Uncertainty Components in Fissile Isotopes and Element Accountancy for the Effective Safeguarding of Nuclear Materials", for three main reasons. The proposed title -

1) refers to "Accountancy" without mentioning the measurements which in fact are the substance of the article
2) refers to "Effective" safeguarding without really addressing, in the article, the way in which the various safeguards measures together make safeguarding effective
3) refers to the plural "Nuclear Materials" which formally is a not defined concept in comparison to the carefully defined and, in INFCIRC/153 enshrined term, "nuclear material".

Abstract

Following a 1988 recommendation of the IAEA Standing Advisory Group on Safeguards Implementation (SAGSI), the IAEA convened a Consultants Group Meeting in June 1991 to provide expert advice on international standards of measurements applicable to safeguards data. As a result, International Target Values (ITVs) were chosen to describe uncertainty components in fissile element and isotope accountancy and were defined on the model of the 1987 and 1988 ESARDA Target Values. The draft ITVs were submitted to several national and international technical panels including ESARDA/WGDA, ESARDA/WGNDA, INMM 5.1 Committee, ISO/TC85/SC5/WG3, and the Japanese Ad Hoc Technical Group on International Standards of Measurements. The results of this world wide review were incorporated in the present report of the IAEA Consultants Group Meeting of June 1993. The 1993 ITVs in this report describe the within- and between-inspection uncertainty components which are expected to be achievable under current industrial and inspection conditions in fissile element and isotope accountancy measurements. Sixteen different nuclear materials of major significance to safeguards are considered. Separate ITVs are defined for the major measurement steps including bulk measurement, sampling, isotope and element assays for NDA and DA procedures. The propagation of these components defines the ITVs for uncertainties in the determinations of masses of fissile element and isotope for selected combinations of measurement methods. These ITVs are directly comparable to the uncertainties observed in actual operator's and inspector's accountancy data. The present report explains why target values are needed, how the concept evolved and how they relate to the operator's and inspector's measurement systems. The 1993 ITVs are intended to be used by plant operators and safeguards organisations, as a reference of the quality of measurements achievable in nuclear material accountancy, and for planning purposes. They should not be used in place of performance values in statistical tests of operator-inspector differences, nor for licensing or regulatory purposes. The report acknowledges the progress made in accountancy and verification measurements since the 1988 ESARDA Target Values were published and points out the areas where further improvements can be expected in the future.

Introduction

Safeguarding nuclear material involves a quantitative verification of the accountancy of fissile materials by independent measurements. The effectiveness of these verifications depends to a great extent upon the quality of the accountancy measurements achieved by both the facility operator and the safeguards inspectorate. For this reason a typical model of Safeguards Agreements /1/ stipulates that: *"The agreements should provide that the system of measurements on which the records used for the preparation of reports are based, shall either conform to the latest standards or be equivalent in quality to such standards"*. Although the above requirement was directed to the facility operators, it applies equally well to the safeguards inspectorates.

In the absence of relevant international standards of measurements, the International Atomic Energy Agency (IAEA) had defined in the 1970s a set of international standards of nuclear material accountancy /2/, which lists the "expected measurement accuracy associated with the closing of a material balance" at five different types of nuclear facilities. However, these values have never been reviewed despite numerous technological changes since their adoption by consensus by a group of experts designated by their Governments. Safeguards officials and evaluators but also plant measurement specialists would need both more current and more informative references regarding the desirable precision and accuracy to be achieved in the measurements of the volume or mass of a material and in the sampling, elemental and isotopic assays for the various nuclear materials encountered in the nuclear fuel cycle.

The Working Group on Techniques and Standards for Destructive Analysis (WGDA) of the European Safeguards Research and Development Association (ESARDA) pioneered the way in 1979 by presenting a list of "Target Values" for the uncertainty components in destructive analytical methods /3/ to the safeguards authorities of Euratom and of IAEA. Revised estimates were prepared in collaboration and published as the 1983 Target Values /4/ after four years of extensive discussion and consultation with and within operators' laboratories and safeguards organizations. The international acceptance of the concept grew further with the next review which involved, besides the ESARDA/WGDA and IAEA, the active participation of the members of two specialized committees of the Institute of Nuclear Materials Management (INMM). The 1987 Target Values, published as a result of this review /5/, define, like the previous editions, the values of the random and systematic error parameters to be aimed for in elemental and isotopic analyses of the most significant types of materials using common destructive analytical methods. The same groups took a new step when they agreed to define with the 1988 edition /6/ the values of the random error parameter to be met in the elemental assays as a result of sampling. Unfortunately, it was not possible at this time to include values for the systematic components in the sampling uncertainties.

The present paper establishes the concept of International Target Values (ITVs) and includes estimates of the random and systematic error uncertainties originating from the measurements of volumes or masses of nuclear materials. The scope of ITVs was also extended to include a consideration of the non-destructive assay methods (NDA) which have won acceptance as accountancy verification tools.

As in earlier publications the values listed in the present paper have been derived from an evaluation of actual measurement data. Three sources of this information were considered. The most relevant and complete set of measurement data comes, without question, from the information gathered by safeguards inspectorates during the statistical evaluation of the measurements reported by the facility operators and the results of independent measurements performed on the same materials by the inspectors /7,8/. Secondly, these data, as shown later, must be complemented or confirmed by an examination of the results of laboratory intercomparisons /9-12/ or measurement quality evaluation programmes /13-27/. Lastly, and whenever possible, the proposed ITVs have been supported on the basis of in-depth analyses of the uncertainties derived from individual measurement processes as published by measurement specialists /28-31/.

It is also important to note that an increasingly broader audience took part in the discussion of the successive versions of the ESARDA and the International Target Values. The ESARDA/WGDA held joint meetings with the ESARDA Working Group on NDA methods (ESARDA/WGNDA) as a part of the discussion of the 1993 version. The IAEA included the topic in its current work plan and held two Consultants Group Meetings /32-33/ with the participation of a representative from a large European reprocessing plant, of Brazilian and Japanese nuclear national authorities along with representatives of ESARDA, INMM, the International Organisation for Standardisation (ISO), the European Community (EC) and IAEA inspectorates. In total close to 500 specialists from nearly 100 laboratories, nuclear plants, and national authorities in about 20 different countries were involved in the discussion of the 1993 ITVs. The ITVs have been endorsed by the ANSI/INMM Committee 5.1 on Analytical Chemistry Laboratory Measurement Control, ISO/TC851/SC5/WG3 on Accountancy Analytical Methods in Nuclear Spent Fuel Reprocessing and the ESARDA Working Groups on DA and NDA. They will be submitted to the IAEA Standing Advisory Group on Safeguards Implementation (SAGSI).

The 1993 ITVs bear a date like the previous ESARDA Target Values, since experience has shown that the quality of measurements varies with time as methods are improved or developed in response to technical or social demands. ITVs also reflect the current understanding of the structure of the uncertainty components in nuclear material accountancy measurements. Accordingly, they are likely to change as this understanding improves or varies.

As with the previous lists, the 1993 ITVs should be achievable from today forward under the conditions normally encountered in typical industrial laboratories or during actual safeguards inspections. They do not represent the ultimately achievable measurement uncertainties which could be obtained under exceptional or ideal laboratory conditions, or with most recently developed methods.

Major changes in the nuclear fuel cycle have occurred since the 1988 edition. Chemical processing and fuel fabrication plants currently operate at or near full capacity. New and larger plants came into operation, multiplying, for example, the throughput by a projected factor of 4 at COGEMA, La Hague (France) /34,35/ or by a factor of 3 at Belgonucléaire in Dessel (Belgium) /36/. The burnup of reprocessed spent fuels and the specific activity of the retrieved plutonium continued to increase, often reaching over 35,000 MW days/ton and 4 TBq/g (110 Ci/g), respectively. At the same time the limits of personnel radiation exposure, and the volume and radioactivity level of the releases to the environment were drastically reduced, sometimes by a factor 10 or more /35,37/. Intense efforts also are being made in the nuclear industry to continue to decrease the production of radioactive wastes.

Considerable resources have been invested for the installation of new analytical facilities with greater radiation protection shielding and the development of analytical procedures with remote and automatic handling /38-40/ which allow more or faster analyses with less radiation exposure and reduced radioactive effluents and wastes. Measurements with instruments like high level neutron coincidence counters (HLNCC), K-edge X-ray absorptiometer and fluorescence analyzers (HKED) are used routinely at the plants by inspectors not only to detect partial defects but also to verify the flow and balance of nuclear materials. These techniques, or similar ones developed for safeguards purposes, also find applications in plant process control. The 1993 ITVs reflect a significant improvement by a factor of 2 to 2.5 in the accuracy achievable in practice in the analyses of uranium products and spent fuel solutions. The 1993 list should therefore still be a motivating goal for beginner laboratories and a reasonable but effective reference for experienced laboratories and safeguards evaluators.

Safeguards Accountancy Verification Measurements

As evident from the title of this paper and its introduction, the principal application of the ITVs will continue to be in safeguards activities. The safeguards verification data also form the main measurement information on which the ITVs are based. A description of the origin of the safeguards data is therefore relevant.

Figure 1 describes the basic measurement scheme followed in safeguards accountancy verifications. For each inspection, *j*, the inspector selects, in

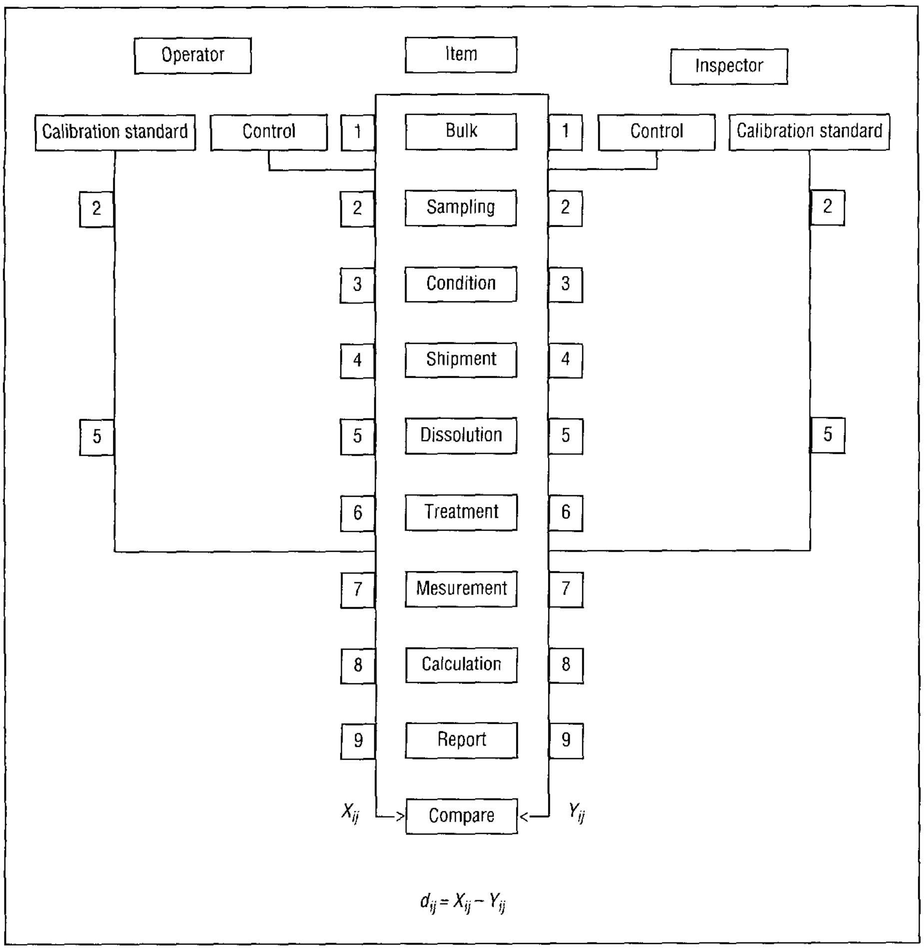

Figure 1: Accountancy and verification measurement scheme for item *i* during inspection *j*

accordance with a random sampling plan, the items or batches of nuclear materials to be verified by an independent measurement. The inspector then compares the result of his verification measurement, Y_{ij}, to the result, X_{ij}, which the operator has obtained on the same batch or item *i*, and which the operator has declared to the inspectorate. The ability of the inspector to detect whether the difference d_{ij} is significantly different from zero depends upon the overall uncertainties in the results X_{ij}, and Y_{ij},. Figure 1 identifies the major steps of the measurement process where uncertainties can arise. Figure 1 reflects more immediately the situation where both the operator and the inspector use a destructive analytical method (DA) for the elemental and isotopic assays. However it is also, with some modification, directly relevant to the use of an NDA method, which is frequently the case for the inspector. Nevertheless, it is recognized that a model more specific to NDA may be needed to take into account differences between NDA and DA methodologies.

Step 1 corresponds to the measurement of the volume or mass of the item or batch of material. This so called *"bulk"* measurement, when needed, takes place in the plant area. Uncertainties due to bulk measurement errors will be considered for the first time in the present paper.

Step 2, the *"sampling"*, involves removing, for the purpose of the analytical measurement, a representative fraction of the material from its container. This step is also done in the plant area.

Step 3 points out the precautions which must be taken in the way the withdrawn sample is *"conditioned"* and packaged at the sampling station so that its isotopic and elemental compositions do not change in an uncontrolled way during its transport to the position or laboratory where it will be measured /41/.

Step 4, the *"shipment"*, is the transport itself of the sample to the location where it can be measured. This is never a trivial operation even when the movement is very short, as in the case of an NDA measurement which is done practically on the spot.

Step 5, the *"dissolution"*, is specific to destructive analytical measurements using a wet chemical procedure.

Step 6, the *"treatment"*, is intended to bring the sample into the optimal geometrical, physical and/or chemical form for the measurement.

Step 7 represents the *"measurement"* itself.

Step 8, the *"calculation"*, involves transforming the result of the physical or chemical measurement obtained in the preceding step into an estimate of the elemental concentration c_{ij} or the isotope abundance f_{ij} of the fissile element or isotope of interest and combining these with the result of the bulk measurement w_{ij} to yield a measure of the mass of fissile isotope X_{ij} in item or batch *i*, or $X_{ij} = w_{ij} \bullet c_{ij} \bullet f_{ij}$. It is in this step that uncertainties in the physical or chemical model used to describe the measurement come into play along with the uncertainties of the calibration process. Figure 1 points out that most often the reference materials used to calibrate the measurement method are, especially for DA, also subject to a sequence of steps before they are measured. This sequence will usually be different and simpler than the process to which the actual sample is subjected. Although not shown on Figure 1, bulk measurements need also to be calibrated. The uncertainties in the measurements of element concentrations and isotope abundances considered in the earlier Target Values and the present version of the ITVs refer to the combined effects of the uncertainties in Steps 3 to 8 occurring after the taking of the sample in Step 2.

Step 9, the *"reporting"* of the results, is purely clerical but unfortunately it can be a source of mistakes. Of course, the goal for Step 9 should be to have no mistakes at all. Thus, no additional uncertainty from the clerical reporting of the results is included in the ITVs proposed in this document.

Quality control should be introduced at every stage of the process, starting with bulk measurements. Quality control on sampling can be done by taking replicate samples after different mixing times or taking samples from a number of items of the same batch of bulk materials. Quality control materials or samples can be introduced at specific steps to verify the quality of the whole process or any part of it, including the conditioning and shipment steps. Figure 1 is an example where control materials are used independently by the operator and the inspector to check the quality of the processes following the bulk measurement. Quality control measures should be documented in a suitable Quality Assurance programme /42-44/.

NOTE: When NDA is used the attention focuses most on the measurement (*Step 7*) as the preceding steps have less impact or may even be omitted. For example, bulk measurements and sampling are not needed if the NDA method allows direct measurement of the total amount of fissile element or isotope contained in a whole item or batch of nuclear material, as with various neutron counters or calorimeters.

Evaluation of Safeguards Accountancy Verification Measurements

The safeguards inspectors examine the operator-inspector data pair differences to determine whether these exceed values which would be expected as a result of the uncertainties in the operators' and inspectors' measurements. The inspectors need to know for this purpose the magnitude of the major uncertainties in the actual data collected during their verification activities.

The two types of measurement uncertainties that play an important role in planning for inspections and in drawing inferences from inspection data stem from the random errors on one side and the systematic errors on the other /45-48/. Simply stated, the effects of random errors are reduced by repeated sampling and analysis while systematic errors continue to persist, and their effects are independent of the number of analyses performed under a fixed set of conditions. Thus, **random error components** can also be understood as those affecting single items. For example, in NDA, single item counting statistics, measured background correction uncertainties are typical random error sources. But also inter-item differences when assaying a population (a batch, a stratum) will appear as a "random" component. **Systematic error components** are those affecting groups or classes of items, like those interpreted with the same calibration curve, or normalized with the same normalization experiments, or affected by the same background subtraction. However uncertainties in the certified values of reference materials used in the calibration curve setting, nuclear data uncertainties, as well as instrumental biases will also appear to have a systematic character.

A systematic error is regarded by some to be synonymous with a calibration error. In some applications, this may in fact be the dominant source of uncertainty. However, calibration information is not always available, nor are shifts in the calibration always taken into account. Further, other sources of systematic errors, resulting, for example, from uncorrected interfering effects, may contribute significantly to the overall uncertainty. Mismatches between actual samples and calibration materials or the settling of powders during storage or shipments can be, for example, the sources of undetected errors in NDA measurements. The resulting uncertainties will usually be expected to have a systematic character. However, under actual inspection conditions these effects can also appear as random uncertainties. Also, in methods using internal calibration procedures, the uncertainty linked to the calibration is essentially of a random rather than systematic nature. Examples of this are the plutonium isotopic analysis by high resolution gamma spectrometry (HRGS) /29/ and isotope dilution mass spectrometry (IDMS) using two isotope tracers /49/.

A basic assumption is that the random and systematic components of the measurement uncertainties are characteristics of the type of material, its chemical and physical form and of the method of measurement. A further assumption is that for a given inspection, the systematic component is constant, but this component may vary from one inspection to another, for both the operator and the inspector. Consequently, the inspectors group the data pairs originating from one inspection, j, by material balance areas (MBA) and by strata of materials of similar characteristics /47/. For a given MBA and stratum, call

$$d_{ij} = \frac{X_{ij} - Y_{ij}}{X_{ij}}$$

and the operator-inspector difference d_{ij} for item i in inspection j, with $i = 1, 2, ..., m_j$ and $j = 1, 2, ..., K$. To simplify the presentation, relative differences are treated here. In practice, absolute differences, $(X_{ij}-Y_{ij})$ are also used. The assumed error model is $d_{ij} = \Delta + \Delta_j + \varepsilon_{ij}$, where Δ is the mean difference over the K inspections, while Δ_j is the change in the systematic error observed during inspection j, and ε_{ij} is the random error affecting the measurement of item i during inspection j.

Δ_j and ε_{ij} both have expectation zero (mean). In a one-way analysis of variance of the operator-inspector differences, d_{ij}, the within-inspection mean square gives an estimate of the variance, $s^2(\varepsilon)$, of the random component, ε_{ij}, while the between-inspection mean square, after removing the contribution of the random component, provides an estimate of the variance, $s^2(\Delta)$, of the changes of the systematic component, Δ_j, from one inspection to another. Alternatively, one may estimate the between-inspection parameter by calculating the variance of the inspection mean differences and correcting for the random component. The model used in the analysis of variance assumes that random uncertainties are the same for all items in the stratum and across all inspections; it assumes also that the between inspection term, Δ_j, has the same probability distribution for all inspections. The reader should consult references /45-48/ for more accurate and detailed descriptions of the model.

Separate paired comparisons of this type are done for bulk measurements, element concentrations and isotope abundances, as well as for the masses of fissile elements and isotopes. After screening out outliers, one obtains, for each type of measurement, an estimate of the sum of the actual variance components for the operator's and inspector's measurement systems:

$$s^2(\varepsilon) = RAN(O)^2 + RAN(I)^2 - 2r(\varepsilon)\, RAN(O) \cdot RAN(I)),$$

and

$$s^2(\Delta) = BIF(O)^2 + BIF(I)^2 - 2r(\Delta)\, BIF(O) \cdot BIF(I)$$

where

$RAN(O)$ and $RAN(I)$ are the standard deviations of the "within-inspection" uncertainty for the operator and the inspector respectively, $r(\varepsilon)$ is the Pearson correlation coefficient between the operator's and the inspector's random errors /50/, and $BIF(O)$ and $BIF(I)$ are the standard deviations of the "*between-inspection-fluctuation*" component, in the operator's and inspector's measurements respectively, and $r(\Delta)$ is the Pearson correlation coefficient between the operator's and inspector's "*between-inspection-fluctuations*" /50/.

Independent evidences show that the uncertainties of operator's and inspector's data have similar magnitudes when both are obtained with similar methods. This is particularly the case when both operator and inspector use DA. In such an instance, the values

$$RAN(O) \cong RAN(I) \cong \frac{s(\varepsilon)}{\sqrt{2}}$$

and

$$BIF(O) \cong BIF(I) \cong \frac{s(\Delta)}{\sqrt{2}}$$

provide good estimates of the standard deviations of the uncertainty components of the measurement of each separate party which are independent of each other, with $r(\varepsilon) = r(\Delta) = 0$. In other situations operator's DA results may be compared with much less precise and/or much less accurate inspector's results obtained for example by some NDA methods. If, for example:

$$RAN(I) \geq 3\ RAN(O),$$

and

$$BIF(I) \geq 3\ BIF(O),$$

then, at the limit

$RAN(I) \cong s(\varepsilon)$ and $BIF(I) \cong s(\Delta)$.

In such a case, $RAN(O)$ and $BIF(O)$ must be derived from a comparison with inspector's measurements obtained by DA. Various other statistical techniques are used to derive separate estimates of the operator's and inspector's uncertainty parameters from a statistical evaluation of the actual data pairs d_{ij} /46-51/. The result of these evaluations are tables of "*Performance*" Values which have been published on the basis of current inspection experience with DA and NDA /53/. These values are generally updated once a year.

Use of Safeguards Performance Values for Inspection Purposes

The Performance Values are used in planning inspections and in drawing inferences based on the declared values of the operator and on the measured values of the inspector. From an inspection planning viewpoint, they allow calculation of sample sizes for

NDA and for DA verification methods that are optimal with respect to achieving the desired level of potential defect detection probability, with the minimum number of samples /54, 55/. With respect to their usage in drawing inferences there are several aspects.

First, they serve to define alarm levels, or reject limits, such that if a given data pair difference, d_{ij}, exceedes the limit L in absolute value, it is identified as a discrepancy, where L is defined by the equation:

$$L = 3\,[s^2(\Delta) + s^2(\varepsilon)]^{1/2}.$$

In practice, the data pair differences may be calculated on either an absolute or relative basis, as was mentioned following the equation in the preceding section. Of course, for homogeneous stratum, it makes no difference whether absolute or relative differences are calculated.

In addition to defining attribute test reject limits as just described, performance values are also used in calculating the standard deviation of the so called $\hat{D}$ statistic which may be applied to a given stratum or to a collection of strata, e.g. all strata comprising an inventory or all strata in a material balance. For a given stratum, $\hat{D}$ is an estimate of the bias in the operator's declared total amount for the stratum, where the bias may be due to a combination of causes, including but not limited to, falsifying data to hide diversion. For a single stratum, de standard deviation of $\hat{D}$ is given by the equation:

$$SD = N\left[s^2(\Delta) + \frac{s^2(\varepsilon)}{n}\right]^{1/2} \bullet \bar{x}$$

where N is the total number of items in the stratum, n is the number of verified items, $\bar{x}$ is the average declared mass of fissile element or isotope per item in the stratum.

In extending $\hat{D}$ to include several strata, the algebraic sum of the individual strata $\hat{D}$ values is found. For example, the estimated bias in an inventory may be detoned by DINV, and the estimated bias in a material balance MUF value by DMUF. The respective deviations, detoned by SDINV and SDMUF in this discussion, are computed as extensions of the above equation, again using performance values. The details of computing SDINV and SDMUF are documented /55a/. It is noted that systematic errors that affect more than one stratum are taken into account. Also, in the case of SDMUF, a distincion is made between the standard deviation of DMUF under the null hypothesis of no data falsification and under the alternate hypothesis of data falsification.

If the inflated variance of DMUF under the alternate hypothesis is ignored to simplify this discussion, then M_o given by equation below is the value of DMUF that would be detected with probability (1-β) if the false alarm probability is α.

$$M_o = (t_\alpha + t_\beta)SDMUF,$$

where t_α and t_β are the normal distribution factors corresponding to a detection probability (1-β) and a risk of false alarm α, respectively.

M_o may be regarded as a measure of the detection capability of the safeguards verification system at the specified probabilities when the verification data are applied in the variable mode. For completeness of discussion, it is noted that one may also combine the test of significance on DMUF and on MUF into a single test involving (MUF- $\hat{D}$). This may be regarded as the inspetor's estimate of MUF since it is the reported MUF corrected for the estimated bias. Again, details of this test are documented /55b/.

Discrepancies in amounts of fissile element or isotope, statistically significant material balance differences exceeding a specified fraction of a safeguards *significant quantity* (SQ) /57/, and SDMUF exceeding a value consistent with the International Standards of Accountancy /2/ are the object of further investigations by the safeguards authorities.

The inspector will also compare the material balances or the MUF declared by the operator with an uncertainty derived from the Performance Values observed in actual operator-inspector differences at the given MBA, when these exist, rather than from the operator's declared performance. Thus, the balance of material declared to be present at the plant in a given stratum at the time of a particular inspection will be compared to a standard deviation, SB, given by the following equation:

$$SB = N\left[\overline{BIF(O)^2} + \frac{\overline{RAN(O)^2}}{n(O)}\right]^{1/2} \bullet \bar{x}$$

where N is the number of items in the material balance, $n(O)$ is the number of items that the operator has measured in this stratum, $BIF(O)$ and $RAN(O)$ are the performance values derived by the inspector from operator-inspector pair differences for the given stratum and facility, and $\bar{x}$ is the average declared mass of fissile element or isotope per item in the stratum.

The estimation of Performance Values from the statistical evaluation of the operator-inspector pair differences constitutes a verification of operator's declared uncertainties. This function of the verification measurement system becomes particularly important when safeguarding very large nuclear material throughputs since *timely detection* of anomalies requires that the operator makes available the process data needed to maintain a *near real time material accountancy* (NRTMA) of the MUF.

Limits of the Safeguards Performance Values

The users of the Performance Values must remain aware of a number of limits in their meaning or content. Plant operational or economic constraints may inflate the variance components of the operator-inspector differences significantly compared to the expected capability of current measurement technology. The safeguards inspector must indeed verify that the uncertainties in the plant measurement system are not deliberately inflated in order to reduce the detection capability of the verification measurements. The latter concern increases with the throughput of the plant. There will therefore always be a need for Target Values providing an accepted measure of the capability of current measurement technology under reasonably economic and operational conditions encountered in the industry. Conversely, paired comparisons do not detect the measurement errors or uncertainties that are common to the operator and inspector. For example, if both use the same reference material for calibration, the uncertainty of the certified value of the reference material will appear as a common systematic component in both results. The common component can also be of a random nature; random sampling errors are common, for instance, when the operator and the inspector measure the same sample or separate aliquots of the same sample.

These common components do not affect the uncertainties of the differences between operator's and inspector's measurements on a single stratum. They can, however, hinder the detection of differences with the true amount of material. On the other hand the use of Performance Values can lead to underestimation of the total uncertainties in the operator's declarations or in the material balance differences over the plant. Independent measurement evidence, free from such common mode uncertainties, is hence needed.

The user of the Performance Values must also know that the estimate of the between inspection effects, s(Δ), becomes less precise as the random uncertainty component, s(ε), increases /46/. When the inspector's uncertainties are large compared to the operator's values, it becomes difficult to obtain a precise estimate of the operator's uncertainties, and vice-versa. This is frequently the case when the operator's data come from DA measurements while the inspector measures by NDA. The paired comparisons can lead to an overestimation of the random uncertainties of the operator's DA measurements, and, at the same time, to a poor estimate of the between-inspection effects in the inspector's NDA results. As a further complication, estimates of these parameters will be affected when the operator's values are based in part on nominal or average values. A separate evaluation of the performance of individual measurement methods is necessary to guard against such potential problems.

Results of Laboratory Intercomparisons

Laboratory intercomparisons also offer a documented set of experimental data of relevance in defining Target Values. The most useful information stems from experiments where the participants analyze very well-characterized materials or measure well-known volumes or masses of nuclear materials in industrial tanks or containers and where their results are directly compared to the certified composition of the materials or to the certified value of the quantities subject of the test. Permanent or periodic measurement evaluation programmes have a greater value for our present purpose than one-shot intercomparison experiments, because the participants tend to follow more closely their routine measurement procedure when the intercomparison samples are submitted sufficiently frequently /14-27/.

Mass measurements are rather straightforward, so that actual inspection data probably provide sufficiently reliable estimates of their uncertainties. The measurements of volumes of solutions in industrial tanks using pneumatic level indicators is a more complex procedure and have been the object of several scientific experiments with international participation. The results of these experiments have been reported /58-61/ and were used in the discussion of the relevant Target Values. The uncertainties to be expected in the use of tracer techniques for volume measurements have been evaluated in the same or similar experiments /62-65/.

Unfortunately, there exists no permanent measurement evaluation programme regarding the quality of sampling procedures. Interlaboratory experiments on the quality of sampling of industrial nuclear materials have been rare and limited to the estimation of the random component of the sampling uncertainties /66/. Actually in most reports of such experiments, the interlaboratory participation concerned more the evaluation of the quality of the elemental assay than the sampling itself.

There are numerous references of interest regarding one-shot intercomparisons of the quality of elemental and isotopic assays by DA /11,12,66-68/. There are yet, however, too few reports of extensive NDA intercomparisons /9,18,70,71/. The evaluations of such one-shot experiments are usually much more elaborate than those of actual inspection data or those of permanent measurement evaluation programmes. They provide, therefore, a better insight into the structure of the sources of measurement uncertainties. Unfortunately, they are often published with long delays and rarely present a current picture. Another frequent drawback of interlaboratory comparisons is that they too rarely involve the measurements of actual industrial materials under industrial conditions. The report of the interlaboratory certification of working reference materials for NDA of plutonium materials /71/ perhaps constitutes an interesting exception.

For these reasons some degree of caution was taken in using the results of these evaluations in the preparation of the present document.

Reports of Validation of Measurement Methods

It is a standard practice of the metrological and analytical laboratories to submit new measurement methods to an experimental validation. However, the reports of such tests have been too frequently overoptimistic and can rarely be checked against independent references. The most trustworthy studies of this type are certainly those which identify the basic metrological parameters of the measurement process, estimate the contributions of the uncertainties occurring in these elementary steps, and compare the expected performance with the results of actual measurements of well-known amounts of materials /28-31,62-65,72-82/.

The reports of most recent experimental developments in isotope dilution mass spectrometric assay of spent fuel solution using Large Size Dried (LSD) spikes /83/, metal spike /84/, internal standard /49/ and total evaporation techniques (TET) /85/ were considered with particular interest because the analyses of spent fuel dissolver solutions at large reprocessing plants should be of the highest possible accuracy.

Meaning of the 1993 International Target Values for Uncertainty Components

The 1993 International Target Values (ITVs) for Uncertainty Components take into account actual practical experiences and should be achievable today under the conditions normally encountered in typical industrial laboratories or during safeguards inspections. They were selected on the basis of a critical discussion of the inspectorates' performance evaluations of actual historical data, and their comparison with the 1987 and 1988 ESARDA Target Values as well as complementary experimental evidence provided by: interlaboratory measurement evaluation programmes, demonstrations of measurement methods and instrumentation, and information provided by individual laboratories.

The 1993 ITVs do not represent the ultimately achievable performance of a measurement system which would be obtained under exceptional or ideal laboratory conditions. However, they reflect reasonably well the progress observed during the past several years in the routine performance of accountancy and verification measurements.

Performance Values are described by a range of values of the parameters measuring the uncertainties observed during *actual* industrial operations and safeguards inspections /52/.

This range is sometimes said to represent the State-of-the-Practice. The uncertainties achieved under *"ideal"* conditions by research laboratories or laboratories producing and certifying primary reference materials can be represented by another range of values which may be taken to illustrate the State-of-the-Art in analytical measurements. At a given time, the two ranges of values can overlap to various degrees depending upon the nature of the measurement and the spread of analytical technology advances at that time. *The ITV for a given type of measurement is a single value which has been selected to be a goal of acceptable level achievable in practice.*

Structure and Content of the 1993 International Target Values

The 1993 ITVs are presented in tabular form according to material types (or strata) which are encountered in nuclear facilities under safeguards and are subject to accountancy measurements.

Tables 1.1 to 1.14 cover fourteen categories of materials of major importance in safeguards. These tables provide the values of significant uncertainty parameters in a measurement for safeguards purposes. They concern primarily the determination of the amounts of uranium (total element), plutonium (total element) and ^{235}U (total isotope), but also provide information on the Target Values proposed for the measurements and processes required to determine total element or isotope amounts, namely bulk measurements, sampling, concentration measurements and the assays of the ^{235}U isotope abundance.

Separate tables provide a list of ITVs for plutonium isotope assays (Table 2) and the codes for the measurement methods used in Tables 1 and 2 (Table 3). Two parameters characterize the quality which should be aimed for in a specific measurement of a given material using a specified method at a *single laboratory*: *RAN* is the relative standard deviation of the repeatability /86/ to be expected in the random uncertainty components encountered during a single inspection, *BIF* is the relative standard deviation of the changes in the systematic errors which may occur between inspections. Attempts were

1993 International Target Values for Uncertainty Components in Measurements of Amount of Nuclear Material for Safeguards Purposes

(% Relative Standard Deviations)

RAN: Relative Standard Deviation of the repeatability of the measurement of a single laboratory within one inspection
BIF: Relative Standard Deviation of the between inspection uncertainty component for a single laboratory

Table 1.1: Material Type: $LEUF_6$

Measurement Method				Bulk		Sampling		U. Conc.		^{235}U Abundance		U Total		^{235}U Total		Pu. Conc.		Pu Total		NOTES
Bulk	U	^{235}U	Pu	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	
LCBS	GRAV	TIMS		0.05	0.05	0.10		0.05	0.05	0.20	0.20	0.10	0.05	0.25	0.20					
LCBS	GRAV	GSMS		0.05	0.05	0.10		0.05	0.05	0.05	0.05	0.10	0.05	0.15	0.10					
		PMCN								5.0	2.0									1/
		PMCG								3.0	2.0									1/

Notes: 1/ Measurement time 300 sec

Table 1.2: Material Type: U Oxide Powders (LEU)

Measurement Method				Bulk		Sampling		U. Conc.		^{235}U Abundance		U Total		^{235}U Total		Pu. Conc.		Pu Total		NOTES
Bulk	U	^{235}U	Pu	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	
EBAL	GRAV	TIMS		0.05	0.05	0.20		0.10	0.10	0.20	0.20	0.25	0.10	0.30	0.25					1/
EBAL	TITR	TIMS		0.05	0.05	0.20		0.10	0.10	0.20	0.20	0.25	0.10	0.30	0.25					1/
EBAL	GRAV	LMCN		0.05	0.05	0.20		0.10	0.10	0.50	0.50	0.25	0.10	0.55	0.50					2/
EBAL	TITR	LMCN		0.05	0.05	0.20		0.10	0.10	0.50	0.50	0.25	0.10	0.55	0.50					2/
		PMCN								2.50	1.50									3/
		PMCG								1.80	1.50									3/, 4/

Notes: 1/ U concentration measurement requires weight change correction because of sample instability
2/ Gamma spectrometry under laboratory conditions
3/ Measurement time 300 sec
4/ Including calibration against reference materials certified to 0.3% or better, and uncertanties in the correction of container wall absorption of 0.5% or less

Table 1.3: Material Type: U Oxide Pellets (LEU)

Measurement Method				Bulk		Sampling		U. Conc.		^{235}U Abundance		U Total		^{235}U Total		Pu. Conc.		Pu Total		NOTES
Bulk	U	^{235}U	Pu	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	
EBAL	GRAV	TIMS		0.05	0.05	0.05		0.05	0.05	0.20	0.20	0.10	0.05	0.20	0.20					3/
EBAL	TITR	TIMS		0.05	0.05	0.05		0.10	0.10	0.20	0.20	0.10	0.10	0.25	0.25					3/
EBAL	GRAV	LMCN		0.05	0.05	0.05		0.05	0.05	0.50	0.50	0.10	0.05	0.50	0.50					1/, 3/
EBAL	TITR	LMCN		0.05	0.05	0.05		0.10	0.10	0.50	0.50	0.10	0.10	0.50	0.50					1/, 3/
		PMCN								2.50	1.50									2/

Notes: 1/ Gamma spectrometry under laboratory conditions
2/ Measurement time 300 sec
3/ The uncertainties in U Total and ^{235}U Total may be larger for pellets containing Gd because of sampling errors due to larger pellet-to-pellet variability

Table 1.4: Material Type: U Scrap

Measurement Method				Bulk		Sampling		U. Conc.		^{235}U Abundance		U Total		^{235}U Total		Pu. Conc.		Pu Total		NOTES
Bulk	U	^{235}U	Pu	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	
EBAL	TITR	TIMS										1.0	0.70	1.5	1.0					1/, 3/
EBAL	TITR	TIMS										5.0	5.0	7.0	7.0					2/, 3/
EBAL	TITR	LMCN										1.0	0.70	1.5	1.0					1/, 3/
EBAL	TITR	LMCN										5.0	5.0	7.0	7.0					1/, 3/
		PMCN								5.0	5.0									4/

Notes: 1/ Homogeneous scrap
2/ Eterogeneous scrap
3/ The values given are representative of average performance observed on historical data. No estimates are given for the individual characteristic; sampling errors are the main contribution to the overall errors observed. Scrap can contain various levels of interfering impurities which could result in larger mesurement errors. NDA measurement not requiring sampling are preferable for heterogeneous scrap
4/ Measurement time 300 sec

Table 1.5: Material Type: Fuel Rods

Measurement Method				Bulk		Sampling		U. Conc.		^{235}U Abundance		U Total		^{235}U Total		Pu. Conc.		Pu Total		NOTES
Bulk	U	^{235}U	Pu	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	
		FRSC												0.70	0.35					1/
		AWCC												2.5	3.5					1/, 3/
		PMCN								3.0	2.0									1/, 4/
			HLNC															2.0	1.0	2/, 3/

Notes:
1/ U Fuel Rods
2/ MOX Fuel Rods
3/ Measurement time 600 sec
4/ Measurement time 300 sec

Table 1.6: Material Type: Fuel Assemblies

Measurement Method				Bulk		Sampling		U. Conc.		^{235}U Abundance		U Total		^{235}U Total		Pu. Conc.		Pu Total		NOTES
Bulk	U	^{235}U	Pu	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	
		UNCL												2.5	2.5					1/
		AWCC												2.0	1.0					1/, 3/
		PMCN								2.0	1.0									1/, 4/
			HLNC															1.5	1.0	2/, 4/

Notes:
1/ U Fuel Assemblies
2/ MOX Fuel Assemblies
3/ Measurement time 600 sec
4/ Measurement time 300 sec

Table 1.7: Material Type: Reprocessing Input Solution (LWR)

Measurement Method				Bulk		Sampling		U. Conc.		^{235}U Abundance		U Total		^{235}U Total		Pu. Conc.		Pu Total		NOTES
Bulk	U	^{235}U	Pu	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	
DIPT	IDMS	TIMS	IDMS	0.30	0.20	0.30		0.20	0.20	0.20	0.20	0.45	0.30	0.50	0.35	0.20	0.20	0.45	0.30	1/
DIPT	HKED		HKED	0.30	0.20	0.30		0.20	0.20			0.45	0.30			0.60	0.30	0.75	0.35	2/

Notes:
1/ U and Pu assay using common spike
2/ Target values for HKED for one-hour counting time

Table 1.8: Material Type: LEU Nitrate, Pu Nitrate, and LEU/Pu Nitrate Solutions

Measurement Method				Bulk		Sampling		U. Conc.		^{235}U Abundance		U Total		^{235}U Total		Pu. Conc.		Pu Total		NOTES
Bulk	U	^{235}U	Pu	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	
DIPT	TITR	TIMS		0.30	0.20	0.10		0.10	0.10	0.20	0.20	0.35	0.20	0.40	0.30					1/
EBAL	TITR	TIMS		0.05	0.05	0.10		0.10	0.10	0.20	0.20	0.15	0.10	0.25	0.25					1/
DIPT	KEDG			0.30	0.20	0.10		0.20	0.20			0.35	0.30							1/,8/
DIPT			TITR	0.30	0.20	0.20										0.15	0.15	0.40	0.25	2/,4/
DIPT			IDMS	0.30	0.20	0.20										0.20	0.20	0.40	0.30	2/
DIPT			KEDG	0.30	0.20	0.20										0.20	0.20	0.40	0.30	2/,5/,8/
DIPT			KEDG	0.30	0.20	0.20										0.60	0.50	0.70	0.55	2/,6/,8/
DIPT	TITR	TIMS	TITR	0.30	0.20	0.20		0.10	0.10	0.20	0.20	0.35	0.20	0.40	0.30	0.20	0.20	0.40	0.30	3/, 4/
DIPT	IDMS	TIMS	IDMS	0.30	0.20	0.20		0.20	0.20	0.20	0.20	0.40	0.30	0.45	0.35	0.20	0.20	0.40	0.30	3/
DIPT	XRF	TIMS	XRF	0.30	0.20	0.20		0.50	0.50	0.20	0.20	0.60	0.55	0.65	0.55	0.50	0.50	0.60	0.55	3/
DIPT	HKED	TIMS	HKED	0.30	0.20	0.20		0.20	0.20	0.20	0.20	0.40	0.30	0.45	0.35	0.60	0.30	0.70	0.35	3/,7/,8/

Notes:
1/ Uranyl nitrate solution
2/ Pu nitrate solutions
3/ Mixed U/Pu solutions with U/Pu ratio between 1 and 100
4/ Coulometry expected to give equivalent performance as potentiometric titration
5/ KEDG with x-ray tube and optical cells
6/ KEDG with radioisotope transmission sources
7/ for LWR-type solutions
8/ Mesurement time 600-1200 sec

Table 1.9: Material Type: Pu Oxide

Measurement Method				Bulk		Sampling		U. Conc.		^{235}U Abundance		U Total		^{235}U Total		Pu. Conc.		Pu Total		NOTES
Bulk	U	^{235}U	Pu	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	
EBAL			GRAV	0.05	0.05	0.10										0.10	0.10	0.15	0.10	
EBAL			TITR	0.05	0.05	0.10										0.15	0.15	0.20	0.15	1/
EBAL			INVS	0.05	0.05	0.10										2.0	1.5	2.0	1.5	2/
			HLNC															1.0	0.50	2/

Notes: 1/ Coulometry expected to give equivalent performance as potentiometric titration
2/ Measurement time 300 sec; with mass spectrometric isotopic analysis

Table 1.10: Material Type: FBR MOX (> 10% Pu)

Measurement Method				Bulk		Sampling		U. Conc.		^{235}U Abundance		U Total		^{235}U Total		Pu. Conc.		Pu Total		NOTES
Bulk	U	^{235}U	Pu	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	
EBAL	TITR	TIMS	TITR	0.05	0.05	0.50		0.20	0.20	0.20	0.20	0.55	0.20	0.60	0.30	0.20	0.20	0.55	0.20	1/, 2/
EBAL	GRAV	TIMS	TITR	0.05	0.05	0.50		0.10	0.15	0.20	0.20	0.50	0.15	0.55	0.25	0.20	0.20	0.55	0.20	1/
EBAL	XRF	TIMS	XRF	0.05	0.05	0.50		0.50	0.50	0.20	0.20	0.70	0.50	0.75	0.55	0.50	0.50	0.70	0.50	
EBAL	IDMS	TIMS	IDMS	0.05	0.05	0.50		0.20	0.20	0.20	0.20	0.55	0.20	0.60	0.30	0.20	0.20	0.55	0.20	
EBAL			INVS	0.05	0.05	0.50										1.95	1.5	2.0	1.5	2/
			HLNC															2.0	1.0	2/, 3/

Notes: 1/ Coulometry expected to give equivalent performance as potentiometric titration
2/ Measurement time 300 sec; with mass spectrometric isotopic analysis
3/ Better measurement performance to be expected for material in standardized containers

Table 1.11: Material Type: LWR MOX (< 10% PU)

Measurement Method				Bulk		Sampling		U. Conc.		^{235}U Abundance		U Total		^{235}U Total		Pu. Conc.		Pu Total		NOTES
Bulk	U	^{235}U	Pu	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	
EBAL	TITR	TIMS	TITR	0.05	0.05	0.50		0.20	0.20	0.20	0.20	0.55	0.20	0.60	0.30	0.20	0.20	0.55	0.20	1/
EBAL	GRAV	TIMS	TITR	0.05	0.05	0.50		0.10	0.15	0.20	0.20	0.50	0.15	0.55	0.25	0.20	0.20	0.55	0.20	1/
EBAL	IDMS	TIMS	IDMS	0.05	0.05	0.50		0.20	0.20	0.20	0.20	0.55	0.20	0.60	0.30	0.20	0.20	0.55	0.20	
EBAL			INVS	0.05	0.05	0.50										1.95	1.50	2.00	1.50	2/
			HLNC															4.00	1.00	2/, 3/

Notes: 1/ Equivalent performance expected for coulometric procedures instead of potentiometric titration
2/ Measurement time 300 sec; with mass spectrometric isotopic analysis
3/ Better measurement performance to be expected for material in standardized containers

Table 1.12: Material Type: MOX Scrap

Measurement Method				Bulk		Sampling		U. Conc.		^{235}U Abundance		U Total		^{235}U Total		Pu. Conc.		Pu Total		NOTES
Bulk	U	^{235}U	Pu	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	
EBAL	TITR	TIMS	TITR									5.0	0.50	5.0	0.55			5.1	0.50	1/
EBAL	IDMS	TIMS	IDMS									5.0	0.50	5.0	0.55			5.0	0.50	1/
EBAL			INVS															7.0	5.0	1/, 2/
			HLNC															7.0	3.0	1/, 2/

Notes: 1/ The values given are representative of average performance observed on historical data. No estimates are given for the individual characteristics; sampling errors are the main contribution to the overall errors observed. Scrap can contain various levels of interfering impurities which could result in larger measurement errors
2/ Measurement time 300 sec

Table 1.13: Material Type: U Metal (HEU)

Measurement Method				Bulk		Sampling		U. Conc.		^{235}U Abundance		U Total		^{235}U Total		Pu. Conc.		Pu Total		NOTES
Bulk	U	^{235}U	Pu	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	
EBAL	GRAV	TIMS		0.05	0.05	0.05		0.05	0.05	0.02	0.02	0.10	0.10	0.10	0.10					
EBAL	TITR	TIMS		0.05	0.05	0.05		0.10	0.10	0.02	0.02	0.10	0.10	0.10	0.10					
EBAL	GRAV	LMCN		0.05	0.05	0.05		0.05	0.05	0.10	0.10	0.10	0.10	0.15	0.15					1/
EBAL	TITR	LMCN		0.05	0.05	0.05		0.10	0.10	0.10	0.10	0.10	0.10	0.15	0.15					1/
		PMCN								0.50	0.50									2/
		PMCG								0.50	0.50									2/

Notes: 1/ Gamma spectrometry under laboratory conditions
2/ Measurement time 300 sec; calibration against reference materials certified to 0.3% or better, and uncertainties in the correction of container wall absorption of 0.5% or less

Table 1.14: Material Type: U-Al (HEU)

Measurement Method				Bulk		Sampling		U. Conc.		^{235}U Abundance		U Total		^{235}U Total		Pu. Conc.		Pu Total		NOTES
Bulk	U	^{235}U	Pu	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	RAN	BIF	
EBAL	TITR	TIMS		0.05	0.05	0.20		0.10	0.10	0.05	0.05	0.25	0.10	0.25	0.10					
EBAL	TITR	LMCN		0.05	0.05	0.20		0.10	0.10	0.20	0.20	0.25	0.10	0.30	0.25					1/
		PMCN								1.0	1.0									2/
		PMCG								1.0	1.0									2/

Notes: 1/ Gamma spectrometry under laboratory conditions

2/ Measurement time 300 sec; calibration against reference materials certified to 0.3% or better, and uncertainties in the correction of container wall absorption of 0.5% or less

made to include in these parameters all uncertainty components which determine the potential difference between the measured and the true value assuming that the operator's and inspector's measurements are completely independent. For example, the values specified for the element and isotope concentration measurements take into account all uncertainties generated in Steps 3 to 8 of Figure 1 and the uncertainties in the reference data and materials used in calibration following the taking of the sample. It has not yet been possible to propose ITVs for the term BIF applicable to sampling.

The combination of the RAN and BIF parameters,

$$s_R = \left(RAN^2 + BIF^2 \right)^{1/2}$$

should be equivalent to the relative standard deviation of the *reproducibility* of the measurement, as it is defined in the relevant ISO Standard /86/, when it is applied to the measurement of a single laboratory.

For a given material and a given combination of DA methods, the ITVs for the mass of element or isotope is equal to the square root of the sum of the squares of the ITVs for the relevant individual components, i.e. bulk measurement, sampling, element concentration and isotope abundance, after rounding to the nearest 0.05 % unit. For example in the first line of Table 1.1, the ITVs for the determination of the mass of ^{235}U isotope are derived as follows:

$$RAN = \left(0.05^2 + 0.10^2 + 0.05^2 + 0.20^2\right)^{1/2} = 0.235$$

rounded to RAN = 0.25; and

$$BIF = \left(0.05^2 + 0.00^2 + 0.05^2 + 0.20^2\right)^{1/2} = 0.212$$

rounded to BIF = 0.20.

Applications of ITVs

The ITVs are values for uncertainties achievable in *routine* measurements involved in the determination of the amount of nuclear materials for materials accountancy and safeguards verification purposes. They are intended to be used as a reference by plant operators, state systems and interna-

Table 2: Plutonium Isotope Assay of PuO_2 and MOX

Isotope Ratio	Methods (1)					
	TIMS		HRGS (2)		LMCA (3)	
	RAN	BIF	RAN	BIF	RAN	BIF
^{238}Pu/^{239}Pu	1.5	1.0	2.0	2.0	1.0	1.0
^{240}Pu/^{239}Pu	0.1	0.05	1.0	1.0	0.7	0.7
^{241}Pu/^{239}Pu	0.2	0.20	1.0	1.0	0.7	0.7
^{242}Pu/^{239}Pu	0.4	0.30	(4)	(4)	(4)	(4)

RAN: Relative Standard Deviation of the repeatability of the measurement of a single laboratory within one inspection

BIF: Relative Standard Deviation of the between inspection uncertainty component for a single laboratory

(1) Typical values for high burn up plutonium

(2) Measurement time 3x100 sec

(3) Measurement time 3x1000 sec with 0.5 g amount of plutonium

(4) The ^{242}Pu/^{239}Pu isotope ratios are not measured by gamma spectrometry but may be estimated by isotopic correlations with relative standard deviations of 5% for the random and between-inspection uncertainty components

Table 3: Coding of Measurement Methods

Measurement	Code	Technique
Bulk	LCBS	Load-Cell Based Weighing System
	EBAL	Electronic Balance
	DIPT	Dip Tubes
U Assay	GRAV	Gravimetry
	TITR	Titration
	IDMS	Isotope Dilution Mass Spectrometry
	KEDG	K-Edge Densitometer
	HKED	Hybrid K-Edge/K-XRF Densitometer
	XRF	X-Ray Fluorescence
Isotopic Analysis	TIMS	Thermal Ionisation Mass Spectrometry
	GSMS	Gas Source Mass Spectrometry
	PMCN	Portable Multichannel Analyzer, NaI-detector
	PMCG	Portable Multichannel Analyzer, GeLi-detector
	LMCN	Laboratory Multichannel Analizer, NaI-detector
	HRGS	High Resolution Gamma Spectrometry
	LMCA	Laboratory Multichannel Analyzer
Total ^{235}U	FRSC	Fuel Rod Scanner
	AWCC	Active Well Coincidence Counter
	UNCL	Uranium Neutron Coincidence Collar
Pu Assay	GRAV	Gravimetry
	TITR	Titration
	COUL	Coulometry
	KEDG	K-Edge Densitometer
	HKED	Hybrid K-Edge/K-XRF Densitometer
	IDMS	Isotope Dilution Mass Spectrometry
	HLNC	High Level Neutron Coincidence Counter
	INVS	Inventory Sample Coincidence Counter

Note: Measurement codes correspond to the codes adopted in the IAEA Safeguards Manual, Part SMO, *SMO 7.1, Annex 1*, IAEA, Vienna (1992-09-30) tional safeguards organisations only. The ITVs are not developed to serve licensing or other regulatory objectives. They should also not be used in place of performance values in estimating the statistical significance of operator-inspector differences or MUF.

The expected relative standard deviation for the total amount of nuclear material may be read directly from the tables for certain combinations of material types, measurement methods and conditions. For example, according to Table 1.7, the overall random uncertainty to be expected for a DA measurement of the total plutonium mass is 0.45 % for one laboratory. The expected relative standard deviation, SDR, of the uncertainties of random character in the difference between operator and inspector estimates of the total plutonium mass by DA in a spent fuel solution should be:

$$SDR = \left(2 \bullet \overline{0.45^2}\right)^{1/2} = 0.64\%$$

It is also possible to calculate the parameters for other combinations of methods from the standard deviations listed in the tables for individual uncertainty components by summing the squares of the components and taking the square root of the sum. If, for example, in Table 1.7, the mass of the input solution is measured rather than its volume, the ITV standard deviations for Pu-total would be calculated by summing the squares of the weighing, sampling and IDMS components as follows:

$$\mathrm{RAN} = (0.05^2 + 0.3^2 + 0.2^2)^{1/2} = 0.36\%;$$

and

$$\mathrm{BIF} = (0.05^2 + 0.2^2)^{1/2} = 0.21\%,$$

assuming RAN = BIF = 0.05% for the weighing of the tank with a load cell. The reader is advised to consult reference /87/ for a description of the application of ITVs to various situations which may be encountered in actual verifications.

Future Developments

It is intended to update the ITV tables regularly in order to incorporate the latest information which may come from inspectorates' performance evaluations based on actual historical data, interlaboratory measurement evaluation programmes, demonstration of new measurement methods and instrumentation, and experimental qualification of recommended sampling procedures /88-98/. Suitable measurement data are needed in particular to define ITVs for the uncertainty component of systematic character in sampling procedures. Models more specific to the NDA measurement processes are being developed by the ESARDA/NDA Working Group to monitor the sources of major uncertainties in actual inspectors' measurements. The experts who took part in the present work will follow attentively how recent developments in bulk measurements /60,61/ and elemental assays /82,85,99/ of spent fuel solutions improve the accuracy of the accountability of large throughputs and inventories of nuclear materials at large plants coming now under safeguards. In preparation of the next revision of the ITVs, the IAEA inspectorate should identify the areas where further improvements would be desirable and possible. The next revision of the ITVs should provide the opportunity to invite more experts from Eastern Europe, South America, Asia, and Africa to participate in the discussion and updating of the International Target Values.

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